UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

GPU NUCLEAR CORPORATION

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 157
License No. DPR-50

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by GPU Nuclear Corporation, et al. (the licensee) dated January 18, 1990 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. DPR-50 is amended by revising paragraph 2.b.(1) as indicated below and by deleting paragraph 2.b.(6).*

   (1) GPU Nuclear Corporation, pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility; and Metropolitan Edison Company, Jersey Central Power and Light Company, Pennsylvania Electric Company to possess the facility in accordance with the procedures and limitations set forth in this license;

3. The license is further amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 157 , are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

4. This license amendment is effective as of its date of issuance, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments:
1. Pages 2 and 5 of license
2. Changes to the Technical Specifications

Date of Issuance: September 25, 1990

*Pages 2 and 5 are attached, for convenience, for the composite license to reflect this change.

f. The owners have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

g. The issuance of this operating license will not be inimical to the common defense and security or to the health and safety of the public;

h. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of Facility Operating License No. DPR-50 is in accordance with 10 CFR Part 50, Appendix D, of the Commission's regulations and all applicable requirements of said Appendix D have been satisfied; and

i. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70, including 10 CFR Section 30.33, 40.32, 70.23 and 70.31.

2. Facility Operating License No. DPR-50 is hereby issued to the Metropolitan Edison Company, Jersey Central Power and Light Company, Pennsylvania Electric Company and GPU Nuclear Corporation to read as follows:

   a. This license applies to the Three Mile Island Nuclear Station, Unit 1, a pressurized water reactor and associated equipment (the facility), owned by the Metropolitan Edison Company, Jersey Central Power and Light Company, Pennsylvania Electric Company and operated by GPU Nuclear Corporation. The facility is located in Dauphin County, Pennsylvania, and is described in the "Final Safety Analysis Report" as supplemented and amended (Amendments 1 through 47) and the Environmental Report as supplemented and amended (Amendments 1 and 2).

   b. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

      (1) GPU Nuclear Corporation, pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility; and Metropolitan Edison Company, Jersey Central Power and Light Company, Pennsylvania Electric Company to possess the facility in accordance with the procedures and limitations set forth in this license;

c. Identification of process sampling points;

d. Procedure for the recording and management of data;

e. Procedures defining corrective actions of off control point chemistry conditions; and

f. A procedure identifying (1) the authority responsible for the interpretation of the data, and (2) the sequence and timing of administrative events required to initiate corrective action.

(6) Inservice Testing - DELETED

(7) Aircraft Movements

Sixty (60) days following the report on aircraft movements at the Harrisburg International Airport for the calendar year 1984 pursuant to Technical Specification 6.9.1.B.2.b, a report shall be submitted updating the aircrash probability analysis presented by Metropolitan Edison Company to the Atomic Safety and Licensing Appeal Board in the Three Mile Island, Unit No. 2 operation license proceeding (Docket No. 50-320). Such report shall utilize current data on aircraft movements at the Harrisburg International Airport and updated national aerial crash rates and shall be based on the same methodology presented by Metropolitan Edison Company as accepted by the Appeal Board in ALAB-692. Following receipt of such report NRC will, after discussion with GPU Nuclear Corporation, determine the need for further periodic aircrash probability analyses.

ATTACHMENT TO LICENSE AMENDMENT NO.

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove	Insert
1-1	1-1
1-2	1-2
1-3	1-3
1-4	1-4
1-7	1-7
2-1	2-1
2-2	2-2
2-3	2-3
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2-6	2-6
2-7	2-7
2-9	2-9
3-2	3-2
3-4	3-4
3-5	3-5
3-6	3-6
3-7	3-7
3-10	3-10
3-11	3-11
3-16	3-16
3-17	3-17

Remove	Insert
4-48	4-48
4-49	4-49
4-50	4-50
4-51	4-51
4-55b	4-55b
4-55c	4-55c
4-55e	4-55e
4-55g	4-55g
4-81	4-81
5-2	5-2
5-3	5-3
5-4	5-4
5-5	5-5
5-7	5-7
6-4	6-4
6-10	6-10
6-11	6-11
6-12	6-12
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1. DEFINITIONS

The following terms are defined for uniform interpretation of these specifications.

1.1 RATED POWER

Rated power is a steady state reactor core output of 2568 MWt.

1.2 REACTOR OPERATING CONDITIONS

1.2.1 COLD SHUTDOWN

The reactor is in the cold shutdown condition when it is subcritical by at least one percent delta k/k and Tavg is no more than 200°F. Pressure is defined by Specification 3.1.2.

1.2.2 HOT SHUTDOWN

The reactor is in the hot shutdown condition when it is subcritical by at least one percent delta k/k and Tavg is at or greater than 525°F.

1.2.3 REACTOR CRITICAL

The reactor is critical when the neutron chain reaction is self-sustaining and Keff = 1.0.

1.2.4 HOT STANDBY

The reactor is in the hot standby condition when all of the following conditions exist:

a. Tavg is greater than 525°F

b. The reactor is critical

c. Indicated neutron power on the power range channels is less than two percent of rated power

1.2.5 POWER OPERATION

The reactor is in a power operating condition when the indicated neutron power is above two percent of rated power as indicated on the power range channels.

1.2.6 REFUELING SHUTDOWN

The reactor is in the refueling shutdown condition when, even with all rods removed, the reactor would be subcritical by at least one percent delta k/k and the coolant temperature at the decay heat removal pump suction is no more than 140°F. Pressure is defined by Specification 3.1.2. A refueling shutdown refers to a shutdown to replace or rearrange all or a portion of the fuel assemblies and/or control rods.

1.2.7 REFUELING OPERATION

An operation involving a change in core geometry by manipulation of fuel or control rods when the reactor vessel head is removed.

1.2.8 REFUELING INTERVAL

Time between normal refuelings of the reactor, not to exceed 24 months without prior approval of the NRC.

1.2.9 STARTUP

The reactor shall be considered in the startup mode when the shutdown margin is reduced with the intent of going critical.

1.2.10 TAVG

TAVG is defined as the arithmetic average of the coolant temperatures in the hot and cold legs of the loop with the greater number of reactor coolant pumps operating, if such a distinction of loops can be made.

1.2.11 HEATUP - COOLDOWN MODE

The heatup-cooldown mode is the range of reactor coolant temperature greater than 200°F and less than 525°F.

1.2.12 STATION, UNIT, PLANT, AND FACILITY

Station, unit, plant, and facility as used in these technical specifications all refer to TMI Unit 1.

1.3 <u>OPERABLE</u>

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

1.4 <u>PROTECTION INSTRUMENTATION LOGIC</u>

1.4.1 INSTRUMENT CHANNEL

An instrument channel is the combination of sensor, wires, amplifiers, and output devices which are connected for the purpose of measuring the value of a process variable for the purpose of observation, control, and/or protection. An instrument channel may be either analog or digital.

1.4.2 REACTOR PROTECTION SYSTEM

The reactor protection system is shown in Figure 7.1-1 of the Updated FSAR. It is that combination of protection channels and associated circuitry which forms the automatic system that protects the reactor by control rod trip. It includes the four protection channels, their associated instrument channel inputs, manual trip switch, all rod drive control protection trip breakers, and activating relays or coils.

1.4.3 PROTECTION CHANNEL

A PROTECTION CHANNEL as shown in Figure 7.1-1 of the Updated FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply units, amplifiers, and bistable modules provided for every reactor protection safety parameter) is a combination of instrument channels forming a single digital output to the protection system's coincidence logic. It includes a shutdown bypass circuit, a protection channel bypass circuit and a reactor trip module.

1.4.4 REACTOR PROTECTION SYSTEM LOGIC

This system utilizes reactor trip module relays (coils and contacts) in all four of the protection channels as shown in Figure 7.1-1 of the Updated FSAR, to provide reactor trip signals for de-energizing the six control rod drive trip breakers. The control rod drive trip breakers are arranged to provide a one-out-of-two-times-two logic. Each element of the one-out-of-two-times-two logic is controlled by a separate set of two-out-of-four logic contacts from the four reactor protection channels.

1.4.5 ENGINEERED SAFETY FEATURES SYSTEM

This system utilizes relay contact output from individual channels arranged in three analog sub-systems and two two-out-of-three logic sub-systems as shown in Figure 7.1-4 of the Updated FSAR. The logic sub-system is wired to provide appropriate signals for the actuation of redundant engineered safety features equipment on a two-of-three basis for any given parameter.

1.4.6 DEGREE OF REDUNDANCY

The difference between the number of operable channels and the number of channels which, when tripped, will cause an automatic system trip.

1.5 INSTRUMENTATION SURVEILLANCE

1.5.1 TRIP TEST

A TRIP TEST is a test of logic elements in a protection channel to verify their associated trip action.

1.5.2 CHANNEL TEST

A CHANNEL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practical to verify OPERABILITY, including alarm and/or trip functions.

1.5.3 CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrumentation channels measuring the same parameter.

1.5.4 CHANNEL CALIBRATION

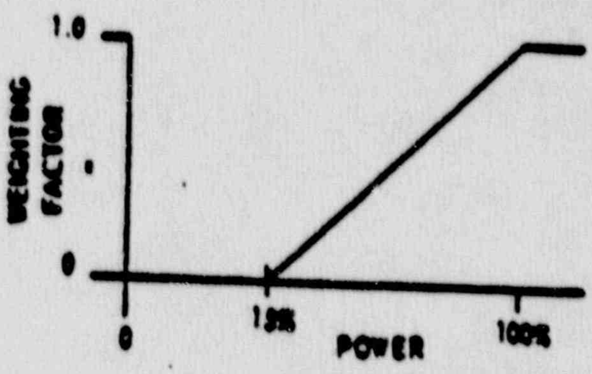
An instrument CHANNEL CALIBRATION is a test, and adjustment (if necessary), to establish that the channel output responds with acceptable range and accuracy to known values of the parameter which the channel measures or an accurate simulation of these values. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include the channel test.

1.5.5 HEAT BALANCE CHECK

A HEAT BALANCE CHECK is a comparison of the indicated neutron power and core thermal power.

1.5.6 HEAT BALANCE CALIBRATION

A HEAT BALANCE CALIBRATION is an adjustment of the power range channel amplifiers output to agree with the core thermal power as defined by a weighted primary and secondary heat balance considering heat losses. The weighting factor, α is shown in the figure below as a function of power level. The equations below define the value of α as a function of power level and the use of α in determining the core thermal power.



1.19 PURGE - PURGING

PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions in such a manner that replacement air o gas is required to purify the confinement.

1.20 VENTING

VENTING is the controlled process of discharging air as gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions in such a manner that replacement air or gas is not provided. Vent used in system name does not imply a VENTING process.

1.21 REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in 10 CFR 50.73.

1.22 MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the GPU System, GPU contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries.

1.23 SUBSTANTIVE CHANGES

SUBSTANTIVE CHANGES are those which affect the activities associated with a document or the document's meaning or intent. Examples of non-substantive changes are: (1) correcting spelling; (2) adding (but not deleting) sign-off spaces; (3) blocking in notes, cautions, etc.; (4) changes in corporate and personnel titles which do not reassign responsibilities and which are not referenced in the Appendix A Technical Specifications; and (5) changes in nomenclature or editorial changes which clearly do not change function, meaning or intent.

1.24 CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT is a TMI-1 specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.5. Plant operation within these operating limits is addressed in individual specifications.

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, axial power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

2.1.1 The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in Figure 2.1-1. If the actual pressure/temperature point is below and to the right of the line, the safety limit is exceeded.

2.1.2 The combination of reactor thermal power and axial power imbalance (power in the top half of core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the safety limit as defined by the locus of points (solid line) for the specified flow set forth in Figure 2.1-2. If the actual-reactor-thermal-power/axial-power-imbalance point is above the line for the specified flow, the safety limit is exceeded.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed, departure from nucleate boiling (DNB). At this point there is a sharp reduction of the heat transfer coefficient, which could result in excessive cladding temperature and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2 (Reference 1) and BWC (Reference 2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark BZ type fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC). A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits.

The curve presented in Figure 2.1-1 represents the conditions at which the minimum allowable DNBR or greater is predicted for the limiting combination of thermal power and number of operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors (Reference 3):

$$F^N_q = 2.82, \; F^N_{\Delta H} = 1.71; \; F^N_z = 1.65$$

The 1.65 cosine axial flux shape in conjunction with $F^N_{\Delta H} = 1.71$ define the reference design peaking condition in the core for operation at the maximum overpower. Once the reference peaking condition and the associated thermal-hydraulic situation has been established for the hot channel, then all other combinations of axial flux shapes and their accompanying radials must result in a condition which will not violate the previously established design criteria on DNBR. The flux shapes examined include a wide range of positive and negative offset for steady state and transient conditions.

These design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion, and form the core DNBR design basis.

The curves of Figure 2.1-2 are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and fuel rod bowing:

a. The DNBR limit produced by a nuclear power peaking factor of $F^N_q = 2.82$ of the combination of the radial peak, axial peak, and position of the axial peak that yields no less than the DNBR limit.

b. The combination of radial and axial peak that prevents central fuel melting at the hot spot. The limit is 20.50 kW/ft.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the axial power imbalance produced by the power peaking.

The specified flow rates for curves 1, 2, and 3 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3. The curves of Figure 2.1-3 represent the conditions at which the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to 22 percent (BAW-2), or 26 percent (BWC) whichever condition is more restrictive.

The maximum thermal power for three pump operation is 89.3 percent due to a power level trip produced by the flux-flow ratio (74.7 per cent flow x 1.08 = 80.6 percent power) plus the maximum calibration and instrumentation error. The maximum thermal power for other reactor coolant pump conditions is produced in a similar manner.

Using a local quality limit of 22 percent (BAW-2), or 26 percent (BWC) at the point of minimum DNBR as a basis for curves 2 and 3 of Figure 2.1-3 is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the BAW-2 or BWC correlation continually increases from the point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

For each curve of Figure 2.1-3, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 (BAW-2) or 1.18 (BWC) or a local quality at the point of minimum DNBR less than 22 percent (BAW-2), or 26 percent (BWC) for the particular reactor coolant pump situation. Curve 1 is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of this curve will be above and to the left of the other curves.

<u>REFERENCES</u>

(1) UFSAR, Section 3.2.3.1.1 - "Fuel Assembly Heat Transfer Design"

(2) BWC Correlation of Critical Heat Flux, <u>BAW-10143P-A</u>, Babcock & Wilcox, Lynchburg, Virginia, <u>April 1985</u>

(3) UFSAR, Section 3.2.3.1.1.3 - "Nuclear Power Factors"

2.2 SAFETY LIMITS - REACTOR SYSTEM PRESSURE

Applicability

Applies to the limit on reactor coolant system pressure

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.

Specification

2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.

Bases

The reactor coolant system (Reference 1) serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME Code, Section III, is 110% of design pressure (Reference 2). The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under ANSI Section B31.7 is 110% of design pressure. Thus, the safety limit of 2750 psig (110% of the 2500 psig design pressure) has been established (Reference 2). The maximum settings for the reactor high pressure trip (2355 psig) and the pressurizer code safety valves (2500 psig) have been established in accordance with ASME Boiler and Pressure Vessel Code, Section III, Article 9, Winter, 1968 to assure that the reactor coolant system pressure safety limit is not exceeded. The initial hydrostatic test was conducted at 3125 psig (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the reactor coolant system pressure does not exceed the safety limit is provided by the presence of a pressurizer electromatic relief valve (Reference 3).

References

(1) UFSAR, Section 4.0 - "Reactor Coolant System"

(2) UFSAR, Section 4.3.10 - "Safety Limits and Conditions"

(3) UFSAR, Table 4.2-8 - "Reactor Coolant System Pressure Settings"

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTION INSTRUMENTATION

Applicability

Applies to instruments monitoring reactor power, axial power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.

Objective

To provide automatic protection action to prevent any combination of process variables from exceeding a safety limit.

Specification

2.3.1 The reactor protection system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3-1 and Figure 2.3-2.

Bases

The reactor protection system consists of four instrument channels to monitor each of several selected plant conditions which will cause a reactor trip if any one of these conditions deviates from a pre-selected operating range to the degree that a safety limit may be reached.

The trip setting limits for protection system instrumentation are listed in Table 2.3-1. These trip setpoints are setting limits on the setpoint side of the protection system bistable comparators. The safety analysis has been based upon these protection system instrumentation trip set points plus calibration and instrumentation errors.

Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

During normal plant operations with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 105.1% of rated power. Adding to this the possible variation in trip set points due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is the value used in the safety analysis (Reference 1).

a. Overpower trip based on flow and imbalance

The power level trip set point produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant flow accident from high power. Analysis has demonstrated that the specified power to flow ratio is adequate to prevent a DNBR of less than 1.30 (BAW-2) or 1.18 (BWC) should a low flow condition exist due to any malfunction.

The power level trip set point produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip set point produced by the power to flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of Table 2.3-1 are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 108 percent and reactor flow rate is 100 percent, or flow rate is 92.5 percent and power level is 100 percent.

2. Trip would occur when three reactor coolant pumps are operating if power is 80.6 percent and reactor flow rate is 74.7 percent or flow rate is 69.4 percent and power level is 75 percent.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 53.1 percent and reactor flow rate is 49.2 percent or flow rate is 45.3 percent and the power level is 49 percent.

The flux/flow ratios account for the maximum calibration and instrumentation errors and the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

No penalty in reactor coolant flow through the core was taken for an open core vent valve because of the core vent valve surveillance program during each refueling outage.

For safety analysis calculations the maximum calibration and instrumentation errors for the power level were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking Kw/ft limits or DNBR limits. The axial power imbalance (power in the top half of the core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio so that the boundaries of Figure 2.3-2 are produced. The power-to-flow ratio reduces the power level trip and associated reactor power/axial power-imbalance boundaries by 1.08 percent for a one percent flow reduction.

b. Pump Monitors

The redundant pump monitors prevent the minimum core DNBR from decreasing below 1.30 (BAW-2) or 1.18 (BWC) by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

c. Reactor coolant system pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip setpoint is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 2.3-1 for high reactor coolant system pressure ensures that the system pressure is maintained below the safety limit (2750 psig) for any design transient (Reference 2). Due to calibration and instrument errors, the safety analysis assumed a 45 psi pressure error in the high reactor coolant system pressure trip setting.

As part of the post-TMI-2 accident modifications, the high pressure trip setpoint was lowered from 2390 psig to 2300 psig. (The FSAR Accident Analysis Section still uses the 2390 psig high pressure trip setpoint.) The lowering of the high pressure trip setpoint and raising of the setpoint for the Power Operated Relief Valve (PORV), from 2255 psig to 2450 psig, has the effect of reducing the challenge rate to the PORV while maintaining ASME Code Safety Valve capability.

A B&W analysis completed in September of 1985 concluded that the high reactor coolant system pressure trip setpoint could be raised to 2355 psig with negligible impact on the frequency of opening of the PORV during anticipated overpressurization transients (Reference 3). The high pressure trip setpoint was subsequently raised to 2355 psig. The potential safety benefit of this action is a reduction in the frequency of reactor trips.

The low pressure (1800 psig) and variable low pressure (11.75 T_{out}-5103) trip setpoint were initially established to maintain the DNB ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction (References 4, 5, and 6). The B&W generic ECCS analysis, however, assumed a low pressure trip of 1900 psig and, to establish conformity with this analysis, the low pressure trip setpoint has been raised to the more conservative 1900 psig. Application of the B&W

f. Shutdown bypass

In order to provide for control rod drive tests, zero power physics testings, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. By administrative control the nuclear overpower trip set point must be reduced to value ≤ 5.0 percent of rated power during reactor shutdown.

2. A high reactor coolant system pressure trip set point of 1720 psig is automatically imposed.

The purpose of the 1720 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The overpower trip set point of ≤ 5.0 percent prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation would be available to remove 5.0 percent of rated power if none of the reactor coolant pumps were operating.

References

(1) UFSAR, Section 1.4.6 - "Criterion 6 - Reactor Core Design"

(2) UFSAR, Section 14.1.2.2 - "Startup Accident"

(3) "Justification for Raising Setpoint for Reactor Trip on High Pressure," BAW-1890, Rev. 0, Babcock and Wilcox, September 1985.

(4) UFSAR, Section 14.1.2.7 - "Stuck-Out, Stuck-In, or Dropped Control Rod Accident"

(5) UFSAR, Section 14.1.2.9 - "Steam Line Break"

(6) UFSAR, Section 14.3, Reference 28 - "ECCS Analysis of B&W's 177-FA Lowered Loop NNS," BAW-10103-A, Rev. 3, Babcock and Wilcox, Lynchburg, Virginia, July 1977.

(7) UFSAR, Section 14.1.2.6 - "Loss of Coolant Flow"

Bases

The limitation on power operation with one idle RC pump in each loop has been imposed since the ECCS cooling performance has not been calculated in accordance with the Final Acceptance Criteria requirements specifically for this mode of reactor operation. A time period of 24 hours is allowed for operation with one idle RC pump in each loop to effect repairs of the idle pump(s) and to return the reactor to an acceptable combination of operating RC pumps. The 24 hours for this mode of operation is acceptable since this mode is expected to have considerable margin for the peak cladding temperature limit and since the likelihood of a LOCA within the 24-hour period is considered very remote.

A reactor coolant pump or decay heat removal pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One decay heat removal pump will circulate the equivalent of the reactor coolant system volume in one-half hour or less.

The decay heat removal system suction piping is designed for 300°F and 370 psig; thus, the system can remove decay heat when the reactor coolant system is below this temperature (References 1, 2, and 3).

Both steam generators must be operable before heatup of the Reactor Coolant System to insure system integrity against leakage under normal and transient conditions. Only one steam generator is required for decay heat removal purposes.

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal or feedwater line break accidents (Reference 4). The pressurizer code safety valve lift set point shall be set at 2500 psig ±1% allowance for error and each valve shall be capable of relieving 280,800 lb/h of saturated steam at a pressure not greater than three percent above the set pressure (Reference 5).

References

(1) UFSAR, Tables 9.5-1 and 9.5-2.

(2) UFSAR, Sections 4.2.5.1 and 9.5 - "Decay Heat Removal"

(3) UFSAR, Section 4.2.5.4 - "Secondary System"

(4) UFSAR, Sections 4.3.10.4 - "System Minimum Operational Components"

(5) UFSAR, Section 4.3.7 - "Overpressure Protection"

Bases

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes (Reference 1). These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4.1-1 of the UFSAR. The maximum unit heatup and cooldown rates satisfy stress limits for cyclic operation (Reference 2). The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100°F satisfies stress levels for temperatures below the Nil Ductility Transition Temperature (NDTT).

The heatup and cooldown rate limits in this specification are based on linear heatup and cooldown ramp rates which by analysis have been extended to accommodate 15°F step changes at any time with the appropriate soak (hold) times. Also, an additional 15°F step change has been included in the analysis with no additional soak time to accommodate decay heat initiation at approximately 252°F.

The unirradiated reference nil ductility temperature (RT NDT) for the surveillance region materials were determined in accordance with 10 CFR 50, Appendixes G and H. For other beltline region materials and other reactor coolant pressure boundary materials, the unirradiated impact properties were estimated using the methods described in BAW-10046A, Rev. 2.

As a result of fast neutron irradiation in the beltline region of the core, there will be an increase in the RT NDT with accumulated nuclear operations. The adjusted reference temperatures have been calculated by adding the predicted radiation-induced RT NDT and the unirradiated RT NDT for each of the reactor coolant beltline materials.

The predicted RT NDT was calculated using the respective neutron fluence after ten effective full power years of operation and the procedures defined in Regulatory Guide 1.99, Rev. 2. The analysis of the reactor vessel material contained in the second Three Mile Island Nuclear Station Unit 1 surveillance capsule confirmed that the current techniques used for predicting the change in impact properties due to irradiation are conservative.

Analyses of the activation detectors in the TMI-1 surveillance capsules have provided estimates of reactor vessel wall fast neutron fluxes for cycles 1 through 4. Extrapolation of reactor vessel fluxes, and corresponding fluence accumulations, based on predicted fuel cycle design conditions during 10 effective full power years of operation are described in Reference 3.

Based on the predicted RT NDT after ten effective full power years of operation, the pressure-temperature limits of Figure 3.1-1 and 3.1-2 have been established in accordance with the requirements of 10 CFR 50, Appendix G. Also, see Reference 4. The methods and criteria employed to establish the operating pressure and temperature limits are as described in BAW-10046A, Rev. 2. The protection against nonductile failure is assumed by maintaining the coolant pressure below the upper limits of these pressure temperature limit curves.

The pressure limit lines on Figures 3.1-1 and 3.1-2 have been established considering the following:

a. A 25 psi error in measured pressure.

b. A 12°F error in measured temperature.

c. System pressure is measured in either loop.

d. Maximum differential pressure between the point of system pressure measurement and the limiting reactor vessel region for the allowable operating pump combinations.

The spray temperature difference restriction, based on a stress analysis of spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

REFERENCES

(1) UFSAR, Section 4.1.2.4 - "Cyclic Loads"

(2) ASME Boiler and Pressure Code, Section III, N-415

(3) BAW-1901, Analysis of Capsule TMI-1C, GPU Nuclear, Three Mile Island Nuclear Station - Unit 1, Reactor Vessel Materials Surveillance Program

(4) BAW-1901, Supplement 1, Analysis of Capsule TMI-1C, GPU Nuclear, Three Mile Island Nuclear Station - Unit 1, Reactor Vessel Materials Surveillance Program, Supplement 1 Pressure - Temperature Limits

3.1.3 MINIMUM CONDITIONS FOR CRITICALITY

Applicability

Applies to reactor coolant system conditions required prior to criticality.

Objective

a. To limit the magnitude of any power excursions resulting from reactivity insertion due to moderator pressure and moderator temperature coefficients.

b. To assure that the reactor coolant system will not go solid in the event of a rod withdrawal or startup accident.

c. To assure sufficient pressurizer heater capacity to maintain natural circulation conditions during a loss of offsite power.

Specification

3.1.3.1 The reactor coolant temperature shall be above 525°F except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply.

3.1.3.2 Reactor coolant temperature shall be above DTT +10°F.

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

3.1.3.4 Pressurizer

3.1.3.4.1 The reactor shall be maintained subcritical by at least one percent delta k/k until a steam bubble is formed and an indicated water level between 80 and 385 inches is established in the pressurizer.

(a) With the pressurizer level outside the required band, be in at least HOT SHUTDOWN with the reactor trip breakers open within 6 hours and be in COLD SHUTDOWN within an additional 30 hours.

3.1.3.4.2 A minimum of 107 kw of pressurizer heaters, from each of two pressurizer heater groups shall be OPERABLE. Each OPERABLE 107 kw of pressurizer heaters shall be capable of receiving power from a 480 volt ES bus via the established manual transfer scheme.

Bases

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods. Calculations show that above 525°F the positive moderator coefficient is acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, startup and operation of the reactor when reactor coolant temperature is less than 525°F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent delta k/k.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient and the small integrated delta k/k would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical below DTT+10°F provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The availability of at least 107 kw in pressurizer heater capability is sufficient to maintain primary system pressure assuming normal system heat losses. Emergency power to heater groups 8 or 9, supplied via a manual transfer scheme, assures redundant capability upon loss of offsite power.

The requirements that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal.

The requirements for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated.

3.1.5 CHEMISTRY

Applicability

Applies to acceptable concentrations of impurities for continuous operation of the reactor.

Objective

To protect the reactor coolant system from the effects of impurities.

Specification

3.1.5.1 If the concentration of oxygen in the primary coolant exceeds 0.1 ppm during power operation, corrective action shall be initiated within eight hours to return oxygen levels to $\leq$ 0.1 ppm.

3.1.5.2 If the concentration of chloride in the primary coolant exceeds 0.15 ppm during power operation, corrective action shall be initiated within eight hours to return chloride levels to $\leq$ 0.15 ppm.

3.1.5.3 If the concentration of fluorides in the primary coolant exceeds 0.10 ppm following modifications or repair to the primary system involving welding, corrective action shall be initiated within eight hours to return fluoride levels to $\leq$ 0.10 ppm.

3.1.5.4 If the concentration limits for oxygen, chloride or fluoride given in 3.1.5.1, 3.1.5.2, and 3.1.5.3 above are not restored within 24 hours of detection, the reactor shall be placed in a hot shutdown condition within 12 hours thereafter. If the normal operational limits are not restored within an additional 24-hour period, the reactor shall be placed in a cold shutdown condition within 24 hours thereafter.

3.1.5.5 If the oxygen, chloride, or fluoride concentration of the primary coolant system exceeds 1.0 ppm the reactor shall be brought to the hot shutdown condition using normal shutdown procedure and action is to be taken to return the system to within normal operation specifications. If normal operating specifications have not been reached in 12 hours, the reactor will then be brought to a cold shutdown condition.

Bases

By maintaining the chloride, fluoride, and oxygen concentration in the reactor coolant within the specifications, the integrity of the reactor coolant system is protected against potential stress corrosion attack (References 1 and 2).

The oxygen concentration in the reactor coolant system is normally expected to be below detectable limits since dissolved hydrogen is used when the reactor is critical. The requirement that the oxygen concentration not exceed 0.1 ppm during power operation is added assurance that stress corrosion cracks will not occur (Reference 3).

If the oxygen, chloride, or fluoride limits are exceeded, measures can be taken to correct the condition (e.g., switch to the spare demineralizer, replace the ion exchange resin, or increase the hydrogen concentration in the makeup tank).

Because of the time dependent nature of any adverse effects arising from chlorides, fluorides, or oxygen concentrations in excess of the limits, and because the condition can be corrected, it is unnecessary to shutdown immediately.

The oxygen, chloride, or fluoride limits specified are at least an order of magnitude below concentrations which could result in damage to materials found in the reactor coolant system even if maintained for an extended period of time (Reference 3). Thus, the period of eight hours to initiate corrective action and the period of 24 hours to perform corrective action to restore the concentration within the limits have been established. The eight hour period to initiate corrective action allows time to ascertain that the chemical analysis are correct and to locate the source of contamination. If corrective action has not been effective at the end of 24 hours, then the reactor coolant system will be brought to the hot shutdown condition within 12 hours thereafter and corrective action will continue. If the normal operational limits are not restored within an additional 24 hour period the reactor shall be placed in cold shutdown condition within 24 hours thereafter.

The maximum limit of 1 ppm for the oxygen, chloride, or fluoride concentration that will not be exceeded was selected because these values have been shown to be safe at 550°F (Reference 4). It is prudent to restrict operation to hot shutdown conditions, if these limits are reached.

REFERENCES

(1) UFSAR, Section 9.2 - "Chemical Addition and Sampling System"

(2) UFSAR, Table 9.2-3 - "Reactor Coolant Quality"

(3) Corrosion and Wear Handbook, D.J. DePaul, Editor

(4) Stress Corrosion of Metals, Logan

3.1.7 MODERATOR TEMPERATURE COEFFICIENT OF REACTIVITY

Applicability

Applies to maximum positive moderator temperature coefficient of reactivity at full power conditions.

Objective

To assure that the moderator temperature coefficient stays within the limits calculated for safe operation of the reactor.

Specification

3.1.7.1 The moderator temperature coefficient shall not be positive at power levels above 95% of rated power.

3.1.7.2 The moderator temperature coefficient shall be $\leq$ + 0.9×10^{-4} delta k/k/F at power levels $\leq$ 95% of rated power.

Bases

A non-positive moderator coefficient (Reference 1) at power levels above 95% of rated power is specified such that the maximum clad temperatures will not exceed the Final Acceptance Criteria based on LOCA analyses. Below 95% of rated power the Final Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of $+0.9 \times 10^{-4}$ delta k/k/F. All other accident analyses as reported in the UFSAR have been performed for a range of moderator temperature coefficients including $+0.9 \times 10^{-4}$ delta k/k/F.

A non-positive moderator coefficient at power levels above 95% of rated power is also required to prevent overpressurization of the reactor coolant system in the event of a feedwater line break (see Specification 2.3.1, Basis C, Reactor Coolant System Pressure).

The Final Acceptance Criteria states that post-LOCA clad temperature will not exceed 2200°F (Reference 2.)

REFERENCES

(1) UFSAR, Section 3.2.2.1.5.4 - "Moderator Temperature Coefficient"
(2) UFSAR, Section 14 - Tables 14.2-1, 14.2-13, 14.2-14

3.1.8 Single Loop Restrictions

Applicability

Applies to single loop operation of the reactor coolant system

Specification

3.1.8.1 Single loop operation while the reactor is critical is prohibited.

Bases

The restriction prohibiting single loop operation with TMI-1 may be lifted, provided that: (1) analyses of TMI-1 support single loop operation, (2) testing on TMI-1 supports the analysis of single loop operation, and (3) any additional equipment necessary for single loop operation is installed at TMI-1.

3.1.9 LOW POWER PHYSICS TESTING RESTRICTIONS

Applicability

Applies to Reactor Protection System requirements for low power physics testing.

Objective

To assure an additional margin of safety during low power physics testing.

Specification

The following special limitations are placed on low power physics testing.

3.1.9.1 Reactor Protection System Requirements

a. Below 1720 psig Shutdown Bypass trip setting limits shall apply in accordance with Table 2.3-1.

b. Above 1800 psig nuclear overpower trip shall be set at less than 5.0 percent. Other settings shall be in accordance with Table 2.3-1.

3.1.9.2 Startup Rate Rod Withdrawal Hold (Reference 1) Shall be operable At All Times.

3.1.9.3 Shutdown margin may not be reduced below 1% delta k/k per 3.5.2.1.

Bases

The above specification provides additional safety margins during low power physics testing, as is also provided for startup (Reference 2.)

REFERENCES

(1) UFSAR, Section 7.2.2.1.b - "Reactivity Rate Limits"

(2) UFSAR, Section 14.1.2.2 - "Startup Accident"

Bases

The safety function enhanced by this venting capability is core cooling. For events beyond the present design basis, this venting capability will substantially increase the plants ability to deal with large quantities of noncondensible gas which could interfere with natural circulation (i.e., core cooling).

The reactor vessel head vent (RC-V42 & RC-V43 in series) provides the capability of venting noncondensible gases from the majority of the reactor vessel head as well as the Reactor Coolant hot legs (to the elevation of the top of the outlet nozzles) and cold legs (through vessel internals leakage paths, to the elevation of the top of the inlet nozzles). This vent is routed to containment atmosphere.

Venting for the pressurizer steam space (RC-V28 and RC-V44 in series) has been provided to assure that the pressurizer is available for Reactor Coolant System pressure and volume control. This vent is routed to the Reactor Coolant Drain Tank.

Additional venting capability has been provided for the Reactor Coolant hot leg high points (RC-V40A, B, RC-41A, B), which normally cannot be vented through the Reactor vessel head vent or pressurizer steam-space vent. These vents relieve to containment atmosphere through a rupture disk (set at low pressure).

The above vent systems are seismically designed and environmentally qualified in accordance with the May 23, 1980 Commission Order and Memorandum per NUREG-0737, Item II.B.1. The high point vents do not fall within the scope of 10 CFR 50.49, since the vents are not relied upon during or following any design basis event (Reference 1). The power operated valves (2 in series in each flow path) which are powered from emergency buses fail closed on loss of power. All vent valves for the reactor vessel head vent, pressurizer vent and loop B high point vent are powered from the class 1E "B" bus. The vent valves for the loop A high point vent are powered from the class 1E "A" bus. The power operated valves are controlled in the Control Room. The individual vent path lines are sized so that an inadvertent valve opening will not constitute a LOCA as defined in 10 CFR 50.46(c)(1). These design features provide a high degree of assurance that these vent paths will be available when needed, and that inadvertent operation or failures will not significantly hamper the safe operation of the plant (Reference 2).

REFERENCES

(1) UFSAR, Section 4.2.3.9 - "Reactor Coolant System Venting"

(2) UFSAR, Section 7.3.2.2.c (16) - "Reactor Coolant System Venting"

3.2 MAKEUP AND PURIFICATION AND CHEMICAL ADDITION SYSTEMS

Applicability
Applies to the operational status of the makeup and purification and the chemical addition systems.

Objective
To provide for adequate boration under all operating conditions to assure ability to bring the reactor to a cold shutdown condition.

Specification
The reactor shall not be critical unless the following conditions are met:

3.2.1 Two makeup and purification pumps are operable except as specified in 3.3.2. Specification 3.0.1 applies.

3.2.2 A source of concentrated boric acid solution, in addition to the borated water storage tank, is available and operable. This can be either:

a. The boric acid mix tank containing at least the equivalent of 906 ft^3 of 8700 ppm boron as boric acid solution with a temperature of at least 10°F above the crystallization temperature. System piping and valves necessary to establish a flow path from the tank to the makeup and purification system shall also be operable and shall have at least the same temperature requirement as the boric acid mix tank. One associated boric acid pump shall be operable.

b. A reclaimed boric acid storage tank containing at least the equivalent of 906 ft^3 of 8700 ppm boron as boric acid solution with a temperature of at least 10°F above the crystallization temperature. System piping and valves necessary to establish a flow path from the tank to the makeup and purification system shall also be operable and shall have at least the same temperature requirement as the reclaimed boric acid tank. One associated reclaimed boric acid pump shall be operable.

c. With neither the boric acid mix tank nor the reclaimed boric acid storage tank OPERABLE, restore one source to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to 1% delta k/k at 200°F; restore a concentrated boric acid source to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

Bases
The makeup and purification system (Reference 1), and the chemical addition and sampling systems (Reference 2) provide control of the reactor coolant boron concentration. This is normally accomplished by using any of the three makeup and purification pumps in series with a boric acid pump associated with the boric acid mix tank or a reclaimed boric acid pump associated with a reclaimed boric acid storage tank. The alternate method of boration will be the use of the makeup and purification pumps taking suction directly from the borated water storage tank (Reference 3).

The quantity of boric acid in storage from either of the three above mentioned sources is sufficient to borate the reactor coolant system to a one percent subcritical margin in the cold condition at the worst time in core life with a stuck control rod assembly. Minimum volumes (including a 10 percent safety factor) of 906 ft^3 of 8700 ppm boron as concentrated boric acid solution in the boric acid mix tank or in a reclaimed boric acid storage tank or 40,000 gallons of 2270 ppm boron as boric acid solution in the borated water storage tank will each satisfy this requirement. Technical Specification 3.3 assures that at least two of these supplies are available whenever the reactor is critical so that a single failure will not prevent boration to a cold condition. The minimum volumes of boric acid solution given include the boron necessary to account for xenon decay.

The primary method of adding boron to the reactor coolant system is to pump the concentrated boric acid solution (8700 ppm boron, minimum) into the makeup tank using either the 10 gpm boric acid pumps or the 30 gpm reclaimed boric acid pumps. Using only one of the two 10 gpm boric acid pumps, the required volume can be injected in less than 13 hours. The alternate method of addition is to inject boric acid from the borated water storage tank using the makeup and purification pumps. The required 40,000 gallons of boric acid can be injected in less than four hours using only one of the makeup and purification pumps.

Concentration of boron in the boric acid mix tank or a reclaimed boric acid storage tank may be higher than the concentration which would crystallize at ambient conditions. For this reason, the boric acid mix tank is provided with an immersion electric heating element and the reclaimed boric acid tanks are provided with low pressure steam heating jackets to maintain the temperature of their contents well above (10°F or more) the crystallization temperature of the boric acid solution contained in them. Both types of heaters are controlled by temperature sensors immersed in the solution contained in the tanks. Further, all piping, pumps and valves associated with the boric acid mix tank and the reclaimed boric acid storage tanks to transport boric acid solution from them to the makeup and purification system are provided with redundant electrical heat tracing to ensure that the boric acid solution will be maintained 10nF or more above its crystallization temperature. The electrical heat tracing is controlled by the temperature of the external surfaces of the piping systems. Once in the makeup and purification system, the boric acid solution is sufficiently well mixed and diluted so that normal system temperatures assure boric acid solubility.

<u>References</u>

(1) UFSAR, Section 9.1,"Makeup and Purification System"

(2) UFSAR, Section 9.2, "Chemical Addition and Sampling Systems"

(3) UFSAR, Figures 6.0-1, 6.0-2 - Simplified ECCS Diagrams

3.3.3 Exceptions to 3.3.2 shall be as follows:

a. Both core flood tanks shall be operable at all times.

b. Both the motor operated valves associated with the core flood tanks shall be fully opened at all times.

c. One reactor building cooling fan and associated cooling unit shall be permitted to be out-of-service for seven days.

3.3.4 Prior to initiating maintenance on any of the components, the duplicate (redundant) component shall be tested to assure operability.

Bases

The requirements of Specification 3.3.1 assure that, before the reactor can be made critical, adequate engineered safety features are operable. Two engineered safeguards makeup pumps, two decay heat removal pumps and two decay heat removal coolers (along with their respective cooling water systems components) are specified. However, only one of each is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident. Both core flooding tanks are required because a single core flooding tank has insufficient inventory to reflood the core for hot and cold line breaks (Reference 1).

The operability of the borated water storage tank (BWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA (Reference 2). The limits on BWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain at least one percent subcritical at 70°F without any control rods in the core following mixing of the BWST and RCS water volumes (Reference 3).

The contained water volume limit of 350,000 gallons includes an allowance for water not usable because of tank discharge location. The limits on contained water volume, NaOH concentration and boron concentration ensure a pH value of between 8.5 and 11.0 of the solution sprayed within containment after a design basis accident. The minimum pH of 8.5 assures that iodine will remain in solution while the maximum pH of 11.0 minimizes the potential for caustic damage to mechanical systems and components. Redundant heaters maintain the borated water supply at a temperature greater than 40°F.

The post-accident reactor building emergency cooling may be accomplished by three emergency cooling units, by two spray systems, or by a combination of one emergency cooling unit and one spray system. The specified requirements assure that the required post-accident components are available.

The iodine removal function of the reactor building spray system requires one spray pump and sodium hydroxide tank contents.

The spray system utilities common suction lines with the decay heat removal system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

When the reactor is critical, maintenance is allowed per Specification 3.3.2 and 3.3.3 provided requirements in Specification 3.3.4 are met which assure operability of the duplicate components. The specified maintenance times are a maximum. Operability of the specified components shall be based on the results of testing as required by Technical Specification 4.5.

An allowable maintenance period of up to 72 hours may be utilized if the operability of equipment redundant to that removed from service is demonstrated immediately prior to removal.

In the event that the need for emergency core cooling should occur, operation of one makeup pump, one decay heat removal pump, and both core flood tanks will protect the core. In the event of a reactor coolant system rupture their operation will limit the peak clad temperature to less than 2,300°F and the metal-water reaction to that representing less than 1 percent of the clad.

Two nuclear service river water pumps and two nuclear service closed cycle cooling pumps are required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant.

<u>REFERENCES</u>

(1) UFSAR, Section 6.1 - "Emergency Core Cooling System"

(2) UFSAR, Section 14.2.2.3 - "Large Break LOCA"

(3) UFSAR, Section 14.2.2.1 - "Fuel Handling Accident"

Bases

A reactor shutdown following power operation requires removal of core decay heat. Normal decay heat removal is by the steam generators with the steam dump to the condenser when RCS temperature is above 250°F and by the decay heat removal system below 250°F. Core decay heat can be continuously dissipated up to 15 percent of full power via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the steam generators is provided by the main feedwater system.

The main steam safety valves will be able to relieve to atmosphere the total steam flow if necessary. Below 5% power, only a minimum number of Main Steam Safety Valves need to be operable as stated in Technical Specification 3.4.1.2.1 and 3.4.1.2.2. This is to provide Steam Generator overpressure protection during hot functional testing and low power physics testing. Additionally, when the Reactor is between hot shutdown and 5% full power operation, the over power trip setpoint in the RPS shall be set to less than 5% as is specified in Technical Specification 3.4.1.2.2. The minimum number of valves required to be operable allows margin for testing without jeopardizing plant safety. Plant specific analysis shows that one Main Steam Safety Valve is sufficient to relieve reactor coolant pump heat and stored energy when the reactor is subcritical by 1% delta K/K for at least one hour. Other plant analyses show that two (2) Main Steam Safety Valves on either OTSG are more than sufficient to relieve reactor coolant pump heat and stored energy when the reactor is below 5% full power operation but had been subcritical by 1% delta K/K for at least one hour since power operation above 5% full power. According to Technical Specification 3.1.1.2a, both steam generators shall be operable whenever the reactor coolant average temperature is above 250°F. This assures that all four (4) Main Steam Safety Valves are available for redundancy. During power operations at 5% full power or above, if Main Steam Safety Valves are inoperable, the power level must be reduced, as stated in Technical Specification 3.4.1.2.3 such that the remaining safety valves can prevent overpressure on a turbine trip.

In the unlikely event of complete loss of off-site electrical power to the station, decay heat removal is by either the steam-driven emergency feedwater pump, or two half-sized motor-driven pumps. Steam discharge is to the atmosphere via the Main Steam Safety Valves and controlled atmospheric relief valves, and in the case of the turbine driven pump, from the turbine exhaust.

Both motor-driven pumps are required initially to remove decay heat with one eventually sufficing. The minimum amount of water in the condensate storage tanks, contained in Technical Specification 3.4.1.1., will allow cooldown to 250°F with steam being discharged to the atmosphere. After cooling to 250°F, the decay heat removal system is used to achieve further cooling.

When the RCS is below 250°F, a single DHR string, or single OTSG and its associated emergency feedwater flowpath is sufficient to provide removal of decay heat at all times following the cooldown to 250°F. The requirement to maintain two OPERABLE means of decay heat removal ensures that a single failure does not result in a complete loss of decay heat removal capability. The requirement to keep a system in operation as necessary to maintain the system subcooled at the core outlet provides the guidance to ensure that steam conditions which could inhibit core cooling do not occur.

Limited reduction in redundancy is allowed for preventive or corrective maintenance on the primary means for decay heat removal to ensure that maintenance necessary to assure the continued reliability of the systems may be accomplished.

As decay heat loads are reduced through decay time or fuel off loading, alternate flow paths will provide adequate cooling for a time sufficient to take compensatory action if the normal means of heat removal is lost.

With the reactor vessel head removed and 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. The BWST with level at 44 feet provides an equivalent reservoir available as a heat sink. Operability of the BWST is to be determined using calculations based on actual plant data or through plant testing at the time the system is to be declared operable. At such times that either of these means is determined to be operable, removal of the redundant or diverse cooling system is permitted.

Following extensive outages or major core off loading, the decay heat generation being removed from the Reactor Vessel is so low that ambient losses are sufficient to maintain core cooling and no other means of heat removal is required. The system is passive and requires no redundant or diverse backup system. Decay heat generation is calculated in accordance with ANSI 5.1-1979 to determine when this situation exists.

An unlimited emergency feedwater supply is available from the river via either of the two motor-driven reactor building emergency cooling water pumps for an indefinite period of time.

The requirements of Technical Specification 3.4.1.1 assure that before the reactor is heated to above 250°F, adequate auxiliary feedwater capability is available. One turbine driven pump full capacity (920 gpm) and the two half-capacity motor-driven pumps (460 gpm each) are specified. However, only one half-capacity motor-driven pump is necessary to supply auxiliary feedwater flow to the steam generators in the onset of a small break loss-of-coolant accident.

<u>REFERENCES</u>

(1) UFSAR, Table 6.1-4 - ECCS "Single Failure Analysis"

(2) UFSAR, 9.5 - Decay Heat Removal System

3.5.1.7.1 Power may be restored through the breaker with the failed trip feature for up to two hours for surveillance testing per T.S. 4.1.1.

3.5.1.8 During STARTUP, HOT STANDBY or POWER OPERATION, in the event that one of the two regulating control rod power SCR electronic trips is inoperable, within one hour:

a. Place the inoperable SCR electronic trip in the tripped condition or
b. Remove the power supplied to the associated SCRs. Specification 3.0.1 applies.

3.5.1.8.1 Power may be restored through the SCRs with the failed electronic trip for up to two hours for surveillance testing per T.S. 4.1.1.

3.5.1.9 The reactor shall not be in the Startup mode or in a critical state unless both HSPS actuation logic trains associated with the Functional units listed in Table 3.5-1 are operable except as provided in Table 3.5-1,D.

3.5.1.9.1 With one HSPS actuation logic train inoperable, restore the train to OPERABLE or place the inoperable device in an actuated state within 72 hours or be in HOT SHUTDOWN within the next 12 hours. With both HSPS actuation logic trains inoperable, restore one train to OPERABLE within 1 hour or be in HOT SHUTDOWN within the next 6 hours.

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless three power range neutron instrument channels and two channels each of the following are operable: four reactor coolant temperature instrument channels, four reactor coolant flow instrument channels, four reactor coolant pressure instrument channels, four pressure-temperature instrument channels four flux-imbalance flow instrument channels, four power-number of pumps instrument channels, and four high reactor building pressure instrument channels. The reactor trip, on loss of feedwater may be bypassed below 7% reactor power. The bypass is automatically removed when reactor power is raised above 7%. The reactor trip, on turbine trip, may be bypassed below 45% reactor power (Reference 1). The safety feature actuation system must have two analog channels functioning correctly prior to startup.

The anticipatory reactor trips on loss of feedwater pumps and turbine trip have been added to reduce the number of challenges to the safety valves and power operated relief valve but have not been credited in the safety analyses.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column "B" (Table 3.5-1). This is in agreement with redundancy and single failure criteria of IEEE 279 as described in the UFSAR Section 7.

There are four reactor protection channels. Normal trip logic is two out of four. Required trip logic for the power range instrumentation channels is two out of three. Minimum trip logic on other instrumentation channels is one out of two.

HSPS instrument operability specified meets the single failure criterion for the EFW system. Four instrument channels are provided for automatic EFW initiation on OTSG low level and high reactor building pressure, and for automatic main feedwater isolation on low OTSG pressure. Normal trip logic is two out of four. With one of the 4 channels in bypass, a second channel may be taken out of service (placed in the tripped position) and no single active failure will prevent actuation of the associated HSPS train actuation logic. No single active failure of either HSPS train will prevent the other HSPS train from operating to supply EFW to both OTSGs.

REFERENCE

(1) B&W Report No. BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," Rev. 0, October 1985

e. If an acceptable axial power imbalance is not achieved within four hours, reactor power shall be reduced until imbalance limits are met.

f. Axial power imbalance shall be monitored on a minimum frequency of once every two hours during power operation above 40 percent of rated power.

3.5.2.8 A power map shall be taken at intervals not to exceed 30 effective full power days using the incore instrumentation detection system to verify the power distribution is within the limits shown in Figure 3.5-2M.

Bases

The axial power imbalance, quadrant power tilt, and control rod position limits are based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5-2M). These limits are developed in a manner that ensures the initial condition LOCA maximum linear heat rate will not cause the maximum clad temperature to exceed 10 CFR 50 Appendix K. Operation outside of any one limit alone does not necessarily constitute a situation that would cause the Appendix K Criteria to be exceeded should a LOCA occur. Each limit represents the boundary of operation that will preserve the Acceptance Criteria even if all three limits are at their maximum allowable values simultaneously. The effects of the APSRs are included in the limit development. Additional conservatism included in the limit development is introduced by application of:

a. Nuclear uncertainty factors

b. Thermal calibration uncertainty

c. Fuel densification effects

d. Hot rod manufacturing tolerance factors

e. Postulated fuel rod bow effects

f. Peaking limits based on initial condition for Loss of Coolant Flow transients.

The incore instrumentation system uncertainties used to develop the axial power imbalance and quadrant tilt limits accounted for various combinations of invalid Self Powered Neutron Detector (SPND) signals. If the number of valid SPND signals falls below that used in the uncertainty analysis, then another system shall be used for monitoring axial power imbalance and/or quadrant tilt.

The 25±5 percent overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Regulating
8	APSR (axial power shaping rod bank)

Control rod groups are withdrawn in sequence beginning with group 1. Groups 5,6 and 7 are overlapped 25 percent. The normal position at power is for group 7 to be partially inserted.

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. As discussed above, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full out position (Reference 1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than: 0.65% delta k/k at rated power. These values have been shown to be safe by the safety analysis of the hypothetical rod ejection accident (Reference 2). A maximum single inserted control rod worth of 1.0% delta k/k is allowed by the rod position limits at hot zero power. A single inserted control rod worth 1.0% delta k/k at beginning of life, hot, zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than 0.65% delta k/k ejected rod worth at rated power.

The plant computer will scan for tilt and imbalance and will satisfy the technical specification requirements. If the computer is out of service, than manual calculation for tilt above 15 percent power and imbalance above 40 percent power must be performed at least every two hours until the computer is returned to service.

Reduction of the nuclear overpower trip setpoint to 60% full power when thermal power is equal to or less than 50% full power maintains both core protection and an operability margin at reduced power similar to that at full power.

During the physics testing program, the high flux trip setpoints are administratively set as follows to assure an additional safety margin is provided:

Test Power	Test Setpoint
0	<5%
15	50%
40	50%
50	60%
75	85%
>75	105.1%

REFERENCES

(1) UFSAR, Section 3.2.2.1.2 - "Reactivity Control Distribution"

(2) UFSAR, Section 14.2.2.2 - "Rod Ejection Accident"

(4) Minimum allowed setting is 3560 v. Maximum allowed setting is 3650 v.

(5) Minimum allowed time is 8 sec. maximum allowed time is 12 sec.

(6) Minimum allowed setting is 2200 volts, maximum allowed setting is 2860 volts

(7) Minimum allowed time is 1.0 second, maximum allowed time is 2.0 seconds.

Bases

High Reactor Building Pressure

The basis for the 30 psig and 4 psig setpoints for the high pressure signal is to establish a setting which would be reached in adequate time in the event of a LOCA, cover a spectrum of break sizes and yet be far enough above normal operation maximum internal pressures to prevent spurious initiation (Reference 1).

Low Reactor Coolant System Pressure

The basis for the 1600 and 500 psig low reactor coolant pressure setpoint for high and low pressure injection initiation is to establish a value which is high enough such that protection is provided for the entire spectrum to break sizes and is far enough below normal operating pressure to prevent spurious initiation. Bypass of HPI below 1775 psig and LPI below 925 psig, prevents ECCS actuation during normal system cooldown (References 1 and 2).

4.16 KV ES Bus Undervoltage Relays

The basis for the degraded grid voltage relay setpoint is to protect the safety related electrical equipment from loss of function in the event of a sustained degraded voltage condition on the offsite power system. The timer setting prevents spurious transfer to the onsite source for transient conditions.

The loss of voltage relay and timers detect loss of offsite power condition and initiate transfer to the onsite source with minimal time delay.

References

(1) UFSAR, Table 7.1-3

(2) UFSAR, Section 14.1.2.10 - "Steam Generator Tube Failure"

2. A comparison check with the incore instrumentation in the event one of the four out-of-core power range detector assemblies gives abnormal readings during operation.

3. Confirmation that the out-of-core axial power splits are as expected.

b. Core power distribution verification includes:

1. Measurement at low power initial reactor startup to check that power distribution is consistent with calculations.

2. Subsequent checks during operation to ensure that power distribution is consistent with calculations.

3. Indication of power distribution in the event that abnormal situations occur during reactor operation.

c. The safety of unit operation at or below 80 percent of operating power for the reactor coolant pump combinations without the core imbalance trip system has been determined by extensive 3-D calculations. This will be verified during the physics startup testing program (Reference 1).

d. The minimum requirement for 23 individual incore detectors is based on the following:

1. An adequate axial imbalance indication can be obtained with nine individual detectors. Figure 3.5-1 shows a typical set of three detector strings with three detectors per string that will indicate an axial imbalance. The three detector strings are the center one, one from the inner ring of symmetrical strings and one from the outer ring of symmetrical strings.

2. Figure 3.5-2 shows a typical detection scheme which will indicate the radial power distribution with 16 individual detectors. The readings from two detectors in a radial quadrant at either plane can be compared with readings from the other quadrants to measure a radial flux tilt.

3. Figure 3.5-3 combines Figures 3.5-1 and 3.5-2 to illustrate a typical set of 23 individual detectors that can be specified as a minimum for axial imbalance determination and radial tilt indication, as well as for the determination of gross core power distributions. Startup testing will verify the adequacy of this set of detectors for the above functions.

e. At least 23 specified incore detectors will be operable to check power distribution above 80 percent power determined by reactor coolant pump combination. These incore detectors will be read out either on the computer or on a recorder. If a set of 23 detectors in specified locations is not operable, power will be decreased to or below 80 percent for the operating reactor coolant pump combination.

REFERENCE

(1) B&W Topical Report No. B&W-10001, "Incore Instrumentation Test Program"

3.5.6 CHLORINE DETECTION SYSTEMS

Applicability

All modes of operation, when chlorine containers exceeding 150 pounds are located onsite.

Objective

To ensure that the Chlorine Detection Systems (CDS) located at the River Water Pump House Chlorinator House and at the Air Intake Structure are capable of providing alarm in the control room and isolating the control room in the event of an onsite chlorine gas release.

Specification

3.5.6.1 Two independent chlorine detection system channels at each of the above locations, shall be Operable. Each channel shall be capable of initiating isolation of the control building ventilation system and providing alarms which allow operators 2 minutes to don emergency breathing apparatus.

Action

3.5.6.2 a. With one chlorine detection system channel at either location inoperable, restore the inoperable channel to OPERABLE status within 7 days. If not restored to OPERABLE status within 7 days, within the next 6 hours, initiate and maintain operation of the control building ventilation system in the emergency recirculation mode of operation.

b. With both chlorine detection system channels at any one location inoperable, within 1 hour initiate and maintain operation of the control building ventilation system in the emergency recirculation mode of operation.

Bases

The Operability of the chlorine detection system ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel (and satisfies the intent of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," Revision 1, January 1977.) (Reference 1)

The Chlorine Detection System is designed so that the human toxicity limits of 15 ppm by volume (45 mg/m³) is not exceeded in the control room within 2 minutes after the operators are made aware of the presence of chlorine.

References

(1) UFSAR, Section 7.4.5.3 - "Toxic Gas Protection"

The hydrogen mixing is provided by the reactor building ventilation system to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

Interspace pressurization leak testing of containment purge valves is performed once every three months. The primary objective of this testing per NRC Safety Issue B-24, is to identify excessive degradation of the resilient seats in a timely manner. Upon failing the quarterly test, manual closure of the valve and retesting are performed in order to identify leakage caused by excessive seat degradation. Manual closure means closure of the valve by means other than the normal operator.

3.8.9 The reactor building purge system, including the radiation monitors which initiate purge isolation, shall be tested and verified to be operable no more than one week prior to refueling operations.

3.8.10 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 72 hours.

Bases

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.7 of the UFSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The decay heat removal pump is used to maintain a uniform boron concentration. The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core (Reference 1). The boron concentration will be maintained above 1800 ppm. Although this concentration is sufficient to maintain the core $k_{eff} \leq 0.99$ if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and replacement. The keff with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

The specification requiring testing Reactor Building purge termination is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Specification 3.8.10 is required as the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours (Reference 2).

REFERENCES

(1) UFSAR, Section 14.2.2.1 - "Fuel Handling Accident"

(2) UFSAR, Section 14.2.2.1(2) - "FHA Inside Containment"

Bases

This specification will limit activity releases to unrestricted areas resulting from damage to spent fuel stored in the spent fuel storage pools in the postulated event of the dropping of a heavy load from the fuel handling building crane. A Fuel Handling accident analysis was performed assuming that the cask and its entire contents of ten fuel assemblies are sufficiently damaged as a result of dropping the cask, to allow the escape of all noble gases and iodine in the gap (Reference 1). This release was assumed to be directly to the atmosphere and to occur instantaneously. The site boundary doses resulting from this accident are 5.25 R whole body and 1.02 R to thyroid, and are within the limits specified in 10 CFR 100.

Specification 3.11.1 requires that spent fuel, having less than 120 days decay post-irradiation, not be loaded in a spent fuel transfer cask in order to ensure that the doses resulting from a highly improbable spent fuel transfer cask drop would be within those calculated above.

Specification 3.11.2 requires the key operated interlock system, which automatically limits the travel area of the fuel handling crane while it is lifting and transporting the spent fuel shipping cask, to be imposed whenever loads in excess of 15 tons are to be lifted and transported while there is any spent fuel in storage in the spent fuel storage pools in Unit 1. This automatically ensures that these heavy loads travel in areas where, in the unlikely event of a load drop accident, there would be no possibility of this event resulting in any damage to the spent fuel stored in the pools, any unacceptable structural damage to the spent fuel pool structure, or damage to redundant trains of safety related components. The shipping cask area is designed to withstand the drop of the spent fuel shipping cask from the 349 ft-0 in. elevation without unacceptable damage to the spent fuel pool structure (Reference 2).

Specification 3.11.3 ensures that the lowest surface of any heavy load never gets higher than one foot above the concrete surface of the 348 ft-0 in. elevation in the fuel handling building (nominal elevation 349 ft-0 in.) thereby keeping any impact force from an unlikely load drop accident within acceptable limits.

Specification 3.11.4 ensures that the proper capacity crane hook is used for lifting and transporting loads thus reducing the probability of a load drop accident.

Following modification or repairs, specification 3.11.5 confirms the load rating of the crane.

References

(1) UFSAR, Section 14.2.2.1 - "Fuel Handling Accident"

(2) UFSAR, Section 14.2.2.8 - "Fuel Cask Drop Accident"

Specification 3.11.6 imposes administrative limits on handling loads weighing in excess of 3000 lbs. to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the spent fuel pool, or to impact redundant safe shutdown equipment. The safe load path shall follow, to the extent practical, structural floor members, beams, etc., such that if the load is dropped, the structure is more likely to withstand the impact. Handling loads of less than 3000 lbs. without these restrictions is acceptable because the consequences of dropping loads in this weight range are comparable to those produced by the fuel handling accident considered in the FSAR and found acceptable.

Specification 3.11.7 in combination with 3.11.3 ensures the spent fuel cask is handled in a manner consistent with the load drop analysis (Reference 3).

Reference

(3) GPU Evaluation of Heavy Load Handling Operations at TMI-1 February 21, 1984, as transmitted to the NRC in GPUN Letter No. 5211-84-2013.

3.14 FLOOD

3.14.1 PERIODIC INSPECTION OF THE DIKES AROUND TMI

Applicability

Applies to inspect on of the dikes surrounding the site.

Objective

To specify the minimum frequency for inspection of the dikes and to define the flood stage after which the dikes will be inspected.

Specification

3.14.1.1 The dikes shall be inspected at least once every six months and after the river has returned to normal, following the condition defined below:

a. The level of the Susquehanna River exceeds flood stage; flood stage is defined as elevation 307 feet at the Nagle Street Bridge, Harrisburg.

Bases

The earth dikes are compacted to provide a stable impervious embankment that protects the site from inundation during the design flood of 1,100,000 cfs. The rip-rap, provided to protect the dikes from wave action and the flow of the river, continues downward into natural ground for a minimum depth of two feet to prevent undermining of the dike (References 1 and 2).

Periodic inspection, and inspection of the dikes and rip-rap after the river has returned to normal from flood stage, will assure proper maintenance of the dikes, thus assuring protection of the site during the design flood.

References

(1) UFSAR, Section 2.6.5 - "Design of Hydraulic Facilities"

(2) UFSAR, Figure 2.6-17 - "Typical Dike Section"

3.14.2 FLOOD CONDITION FOR PLACING THE UNIT IN HOT STANDBY

Applicability

Applies to the river stage for placing the unit in hot standby.

Objective

To define the action taken in the event river elevation reaches 302 feet at the intake structure.

Specification

3.14.2.1 If the river stage reaches elevation 302 feet at the River Water Intake Structure, corresponding to 1,000,000 cfs river flow, the unit will be brought to the hot standby condition.

Bases

The dikes provided protect the plant site during the design flood of 1,100,000 cfs. The design flood corresponds to an elevation of approximately 303 feet at the River Water Intake Structure (Reference 1). The dike elevation at the intake structure is 305 feet. The minimum freeboard is at the downstream end of the plant site where the dike elevation is 304 feet providing a freeboard of approximately one foot. Adequate freeboard is provided to protect the plant site from flooding due to wave action during the design flood (Reference 2).

Placing the unit in hot standby when the river stage reaches 302 feet elevation provides an additional margin of conservatism by assuring that adequate freeboard exists during operation of the unit.

References

(1) UFSAR, Figure 2.6-15 - "Dike Freeboard - Design Flood"

(2) UFSAR, Section 2.6.4 - "Flood Studies"

3.15.2 REACTOR BUILDING PURGE AIR TREATMENT SYSTEM

Applicability

Applies to the reactor building purge air treatment system and its associated filters.

Objective

To specify minimum availability and efficiency for the reactor building purge air treatment system and its associated filters.

Specification

3.15.2.1 Except as specified in Specification 3.15.2.3 below, the Reactor Building Purge Air Treatment System filter AH-F1 shall be operable as defined by the Specification below at all times when containment integrity is required unless the Reactor Building purge isolation valves are closed.

3.15.2.2 a. The results of the in-place DOP and halogenated hydrocarbon tests at maximum available flows on HEPA filters and charcoal adsorber banks for AH-F1 shall show less than 0.05% DOP penetration and less than 0.05% halogenated hydrocarbon penetration; except that the DOP test will be conducted with prefilters installed.

b. The results of laboratory carbon sample analysis for the reactor building purge system filter carbon shall show greater than or equal to 90% radioactive methyl iodide decontamination efficiency when tested at 250°F, 95% R.H.

3.15.2.3 From and after the date that the filter AH-F1 in the reactor building purge system is made or found to be inoperable as defined by Specification 3.15.2.2 above, the Reactor Building purge isolation valves shall be closed until the filter is made operable.

Bases

The Reactor Building Purge Exhaust System (Reference 1) filter AH-F1 while normally used to filter all reactor building exhaust air. It is necessary to demonstrate operability of these filters to assure readiness for service if required to mitigate a fuel handling accident (Reference 2) in the Reactor Building and to assure that 10CFR50 Appendix I limits are met. Reactor Building purging is required to be terminated if the filter is not operable.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal absorbers to prevent clogging of the iodine absorbers for all emergency air treatment systems. The charcoal absorbers are installed to reduce the potential release of radio-iodine to the environment. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10 CFR 100 guidelines for the Fuel Handling Accident which assumes 90% efficiency for inorganic iodines and 70% efficiency for organic iodines.

The flow through AH-F1 can vary from 0 CFM to 50,000 CFM, the maximum purge flow rate During all modes except COLD SHUTDOWN, the purge valves are limited to no more than 30° open (90° being full open). This provides greater assurance of containment isolation dependability per NUREG 0737 Item II.E.4.2 Attachment 1 Item (2)(a). Makeup air is provided between the filter AH-F1 and the fans AH-E7A and B. (See also T.S. 3.6).

The in-place DOP and halogenated hydrocarbon tests of the filter banks and the laboratory tests of the carbon samples will be done using the test methods and acceptance criteria of Regulatory Guide 1.52 (Rev. 2), except that DOP and Freon tests will be performed such that radiation release limitations are not exceeded.

<u>References</u>

(1) UFSAR Section 5.3.3 - "Reactor Building Purge System Isolation"

(2) UFSAR Section 14.2.2.1 - "Fuel Handling Accident"

Bases

The Auxiliary and Fuel Handling Building Air Treatment System (Reference 1) is considered to be the 4 banks of exhaust filters (AH-F2A, B, C, and D) and the two sets of redundant exhaust fans (AH-E-14A and C or AH-E14B and D) which take the exhaust air from both the Auxiliary Building and the Fuel Handling Buildi.g and discharge it to the Auxiliary and Fuel Handling Building exhaust stack. Exhaust air passes through all of the exhaust filters (AH-F2A, B, C, and D) prior to being discharged to the stack whenever either set of AH-E14 exhaust fans is in operation.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers for all emergency air treatment systems. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment.

If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10 CFR 100 guidelines for the accidents analyzed in Chapter 14 of the UFSAR (References 2 and 3), which assumes 90% efficiency. Mitigation of Fuel Handling Accidents is provided by the Fuel Handling Building ESF Air Treatment System and does not depend on the operation of the Auxiliary and Fuel Handling Building Air Treatment System. The Auxiliary and Fuel Handling Building Air Treatment System is isolated by automatic damper actuation in the event of increasing activity in the Fuel Handling Building as sensed by radiation monitors.

References

(1) UFSAR Figure 9.8-4 - "Ventilation Auxiliary & Fuel Handling Building"

(2) UFSAR Section 14.2.2.5 - "Maximum Hypothetical Accident"

(3) UFSAR Section 14.2.2.6 - "Waste Gas Tank Rupture"

3.15.4 Fuel Handling Building ESF Air Treatment System

Applicability

Applies to the Fuel Handling Building (FHB) ESF Air Treatment System and its associated filters.

Objective

To specify minimum availability and efficiency for the FHB ESF Air Treatment System and its associated filters for irradiated fuel handling operations.

Specifications

3.15.4.1 Prior to fuel movement each refueling outage, two trains shall be operable. One train shall be operating continuously whenever TMI-1 irradiated fuel handling operations in the FHB are in progress.

a. With one train inoperable, irradiated fuel handling operations in the Fuel Handling Building may continue provided the redundant train is operating.

b. With both trains inoperable, handling of irradiated fuel in the Fuel Handling Building shall be suspended until such time that at least one train is operable and operating. Any fuel assembly movement in progress may be completed.

3.15.4.2 A FHB ESF Air Treatment System train is operable when its surveillance requirements are met and:

a. The results of the in-place DOP and halogenated hydrocarbon tests at design flows on HEPA filters and carbon adsorber banks shall show <0.05% DOP penetration and <0.05% halogenated hydrocarbon penetration.

b. The results of laboratory carbon sample analysis shall show $\geq$90% radioactive methyl iodide decontamination efficiency when tested at 30°C, 95% R.H.

c. The fans AH-E-137A and B shall each be shown to operate within $\pm$10% of design flow (6,000 SCFM).

Bases

Compliance with these specifications satisfies the condition of operation imposed by the Licensing Board as described in NRC's letter dated October 2, 1985, item 1.c.

The FHB ESF Air Treatment System contains, controls, mitigates, monitors and records radiation release resulting from a TMI-1 postulated spent fuel accident in the Fuel Handling Building as described in the FSAR. Offsite doses will be less than the 10 CFR 100 guidelines for accidents analyzed in Chapter 14 (Reference 1).

Bases (Continued)

Normal operation of the FHB ESF Air Treatment System will be during TMI-1 irradiated fuel movements in the Fuel Handling Building. The system includes air filtration and exhaust capacity to ensure that any radioactive release to atmosphere will be filtered and monitored. Effluent radiation monitoring and sampling capability are provided.

References

(1) UFSAR, Section 14.2.2.1 - "Fuel Handling Accident"

3.17 REACTOR BUILDING AIR TEMPERATURE

Applicability

This specification applies to the average air temperature of the primary containment during power operations.

Objective

To assure that the temperatures assumed in the structural analysis of the Reactor Building are not exceeded.

Specification

3.17.1 Primary containment average air temperature above Elev. 320 shall not exceed 130°F and average air temperature below Elev. 320 shall not exceed 120°F.

3.17.2 If, while the reactor is critical, the above stated temperature limits are exceeded, the average temperature shall be reduced to the above limits within 8 hours, or be in at least HOT STANDBY within the next six (6) hours and in COLD SHUTDOWN within the following thirty (30) hours.

3.17.3 The primary containment average air temperature shall be calculated as follows:

a) The average temperature above elevation 320 will be calculated by taking the arithmetic average of the temperatures from at least 13 locations above elevation 320. A list of locations is given below.

b) The average temperatures below elevation 320 will be calculated by taking the arithmetic average of the temperatures from at least 4 locations below Elev. 320 A list of locations is given below.

Location	Location
SE Wall Elev. 352'	NE Wall Elev 314'*
NW Sec Shield Elev 352'	S Wall Elev 314'*
NE Sec Shield Elev 352'	NW Wall Elev 314'*
E Wall Elev 382'	E Sec Shield Elev 352'
NE Sec Shield Elev 352'	S Rx Wall Elev 321'
NW Sec Shield Elev 352'	NE Wall Elev 287'*
NE Sec Shield Elev 352'	S Wall Elev 287'*
NW Sec Shield Elev 352'	NW Wall Elev 287'*
NW Wall Elev 352'	E Sec Shield Elev 352'
E Wall Elev 400'	NW Sec Shield Elev 287'*
S Sec Shield Elev 352'	NE Sec Shield Elev 364'
NW Sec Shield Elev 352'	N Sec Shield Elev 364'

NOTE: (1) * Detectors located below elev 320'.

TABLE 3.21-1 (continued)

TABLE NOTATION

ACTION 18 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue, provided that prior to initiating a release:

1. At least two independent samples are analyzed in accordance with Specifications 4.22.1.1A & B and;

2. At least two technically qualified members of the Unit staff independently verify the release rate calculations and verify the discharge valve lineup.

3. Director Operations and Maintenance Unit 1 shall approve each release.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 20 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may commence or continue provided that grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least $1x10^{-7}$ microcuries/ml, prior to initiating a release and at least once per 12 hours during release.

ACTION 21 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, radioactive effluent releases via this pathway may continue, provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

3.21.2 RADIOACTIVE GASEOUS PROCESS AND EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.21.2 The radioactive gaseous process and effluent monitoring instrumentation channels shown in Table 3.21-2 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.22.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: As shown in Table 3.21-2.

ACTION:

a. With a radioactive gaseous process or effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, immediately suspend the release of radioactive effluents monitored by the affected channel or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous process or effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.21-2. Exert best efforts to return the instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semi-Annual Effluent Release Report why the inoperability was not corrected in a timely manner.

BASES

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approved methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20.

The low range condenser offgas noble gas activity monitors also provide data for determination of steam generator primary to secondary leakage rate. Channel operability requirements are based on an ASLB Order No. LBP-84-47 dated October 31, 1984, and as cited in 20 NRC 1405 (1984).

TABLE 3.21-2 (Continued)

RADIOACTIVE GASEOUS PROCESS AND EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
5. Auxiliary and Fuel Handling Building Ventilation System			
a. Noble Gas Activity Monitor (RM-A8) or (RM-A4 and RM-A6)	1	*	27
b. Iodine Sampler (RM-A8) or (RM-A4 and RM-A6)	1	*	31
c. Particulate Sampler (RM-A8) or (RM-A4 and RM-A6)	1	*	31
d. Effluent System Flow Rate Measuring Devices (FR-151, or FR-149 and FR-150)	1	*	26
e. Sampler Flow Rate Monitor	1	*	26
6. Fuel Handling Building ESF Air Treatment System			
a. Noble Gas Activity Monitor (RM-A14 or Suitable Equivalent)	1	****	27, 33
b. Iodine Cartridge	N/A(3)	****	31, 33
c. Particulate Filter	N/A(3)	****	31, 33
d. Effluent System Flow (UR-1104A/B)	1	****	26, 33
e. Sampler Flow Rate Monitor	1	****	26, 33

3.22.2.5 The concentration in the Waste Gas Holdup System shall be limited to less than or equal to 2% by volume whenever the concentration of hydrogen in the Waste Gas Holdup System is greater than or equal to 4% by volume.

AVAILABILITY: At all times.

Action: Whenever the concentration of hydrogen in the Waste Gas Holdup System is greater than or equal to 4% by volume, and:

a. The concentration of oxygen in the Waste Gas Holdup System is greater than 2% by volume, but less than 4% by volume, without delay begin to reduce the oxygen concentration to within its limit.

b. The concentration of oxygen in the Waste Gas Holdup System is greater than or equal to 2% by volume, immediately suspend additions of waste gas to the Waste Gas Holdup System and without delay begin to reduce the oxygen concentration to within its limit.

BASES:

Based on experimental data (Reference 1), lower limits of flammability for hydrogen is 5% and for oxygen is 5% by volume. Therefore, if the concentration of either gas is kept below it lower limit, the other gas may be present in higher amounts without the danger of an explosive mixture. Maintaining the concentrations of hydrogen and oxygen such that an explosive mixture does not occur in the waste gas holdup system provides assurance that the release of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10.

REFERENCES:

(1) Bulletin 503, Bureau of Mines; Limits of Flammability of Gases and Vapors.

3.24 Reactor Coolant Inventory Trending System (RCITS)

Applicability

Applies to the operability requirements for the Reactor Coolant Inventory Trending System (RCITS) when the reactor is critical.

Objectives

To assure operability of RCITS instrumentation which may be useful in diagnosing situations which could represent or lead to inadequate core cooling.

Specification (See Note Below)

3.24.1 Reactor Coolant Inventory Trending System (RCITS)

(a) One channel of the two channels shall be OPERABLE. A channel is composed of a hot leg level and a RV head level. With no channels OPERABLE, operation may continue and immediately initiate corrective action to return at least one channel to OPERABLE status as soon as possible. If at least one channel is not restored within 30 days, details shall be provided in the Monthly Operating Report. These details shall include cause, action being taken and projected date for return to OPERABLE status.

(b) One void fraction channel in each Reactor Coolant loop shall be OPERABLE. With no channels OPERABLE, operation may continue and immediately initiate corrective action to return at least one channel to OPERABLE status as soon as possible. If at least one channel is not restored within 30 days, details shall be provided in the Monthly Operating Report. These details shall include cause, action being taken and projected date for return to OPERABLE status.

Bases

The RCS Inventory Trending System (RCITS) (Reference 1) provides indication of the trend in water inventory in the hot legs and reactor vessel during the approach to inadequate core cooling (ICC). In this manner RCITS provides additional information to the operator to diagnose the approach of ICC and to assess the adequacy of responses taken to restore core cooling.

NOTE: This specification is approved only to the end of Cycle 8 of operation. At that time, an amendment must be proposed by GPUN that is consistent with the Standard Technical Specification (STS) resulting from the NRC-initiated Technical Specification Improvement Program.

The RCITS is required to be operable (as defined previously) when the plant is critical.

The RCITS is an information system to aid the operator during the approach to inadequate core cooling. There is no regulatory limit for this system.

Inoperability of the RCITS removes the availability of an information system. Other useful instrumentation for inadequate core cooling will be available.

The RCITS is not a required system to mitigate evaluated accidents. It may be useful to have the RCITS operable but will not cause an adverse impact if it is not operable.

The LCO action statement provides the level of emphasis required for an information system. This allows the plant to continue to operate and not to force an unneeded shutdown.

Reference

(1) UFSAR, Update Section 7.3.2.2(c)10(d) - "Reactor Coolant Inventory Tracking System"

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers shall be checked and calibrated if necessary, every shift against a heat balance standard. The frequency of heat balance checks will assure that the difference between the out-of-core instrumentation and the heat balance remains less than 4%.

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptance tolerances if recalibration is performed at the intervals of each refueling period.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies set forth are considered acceptable.

Testing

On-line testing of reactor protection channels is required monthly on a rotational basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel (Reference 1).

The rotation schedule for the reactor protection channels is as follows:

a) Channels A, B, C & D Before Startup, when shutdown greater than 24 hours and

b) Monthly with one channel being tested per week on a continuous sequential rotation.

The reactor protection system instrumentation test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action in a channel, the instrumentation associated with the protection parameter failure will be tested in the remaining channels. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protection channels coincidence logic, the control rod drive trip breakers and the regulating control rod power SCRs electronic trips, are trip tested monthly. The trip test checks all logic combinations and is to be performed on a rotational basis. The logic and breakers of the four protection channels and the regulating control rod power SCRs shall be trip tested prior to startup when the reactor has been shutdown for greater than 24 hours.

Discovery of a failure that prevents trip action requires the testing of the instrumentation associated with the protection parameter failure in the remaining channels.

For purposes of surveillance, reactor trip on loss of feedwater and reactor trip on turbine trip are considered reactor protection system channels.

The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the equipment and systems in a safe operational status.

REFERENCE

(1) UFSAR, Section 7.1.2.3(d) - "Periodic Testing and Reliability"

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
8.	High Reactor Coolant Pressure Channel	S	M	R	
9.	Low Reactor Coolant Pressure Channel	S	M	R	
10.	Flux-Reactor Coolant Flow Comparator	S	M	R	
11.	(Deleted)	---	---	---	
12.	Pump Flux Comparator	S	M	R	
13.	High Reactor Building Pressure Channel	S	M	R	
14.	High Reactor Building Logic Channels	NA	Q	NA	
15.	High Pressure Injection Analog Channels				
	a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or Tav is greater than 200°F
16.	Low Pressure Injection Logic Channel	NA	Q	NA	
17.	Low Pressure Injection Analog Channels				
	a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or Tav is greater than 200°F
18.	Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	Q	NA	

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
19. Reactor Building Emergency Cooling and Isolation System Channels				
a. Reactor Building 4 psig Channels	S(1)	M(1)	R	(1) When CONTAINMENT INTEGRITY is required
b. RCS Pressure 1600 psig	S(1)	M(1)	NA	(1) When RCS Pressure >1800 psig
c. RPS Trip	S(1)	M(1)	NA	(1) When CONTAINMENT INTEGRITY is required
d. Reactor Bldg. 30 psig	S(1)	M(1)	R	(1) When CONTAINMENT INTEGRITY is required
e. Reactor Bldg. Purge Line High Radiation (AH-V-1A/D)	W(1)	M(1)	R	(1) When CONTAINMENT INTEGRITY is required
f. Line Break Isolation Signal (ICCW & NSCCW)	W(1)	M(1)	R	(1) When CONTAINMENT INTEGRITY is required
20. Reactor Building Spray System Logic Channel	NA	Q	NA	
21. Reactor Building Spray System Analog Channels				
a. Reactor Building 30 psig Channels	NA	M	R	
22. Pressurizer Temperature Channels	S	NA	R	
23. Control Rod Absolute Position	S(1)	NA	R	(1) Check with Relative Position Indicator
24. Control Rod Relative Position	S(1)	NA	R	(1) Check with Absolute Position Indicator
25. Core Flooding Tanks				
a. Pressure Channels	S(1)	NA	R	(1) When Reactor Coolant system pressure is greater than 700 psig
b. Level Channels	S(1)	NA	R	
26. Pressurizer Level Channels	S	NA	R	
27. Makeup Tank Level Channels	D(1)	NA	R	(1) When Makeup and Purification System is in operation

Bases

Specifications 4.2.1 and 2 ensure that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a(g). Relief from any of the above requirements has been provided in writing by the NRC and is not a part of these technical specifications.

4.3 TESTING FOLLOWING OPENING OF SYSTEM

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To assure Reactor Coolant System integrity prior to return to criticality following normal opening, modification, or repair.

Specification

4.3.1 When Reactor Coolant System repairs or modifications have been made, these repairs or modifications shall be inspected and tested to meet all applicable code requirements prior to the reactor being made critical.

4.3.2 Following any opening of the Reactor Coolant System, it shall be leak tested at not less than 2285 psig prior to the reactor being made critical.

4.3.3 The limitations of Specification 3.1.2 shall apply.

Bases

Repairs for modifications made to the Reactor Coolant System are inspectable and testable under applicable codes, such as B 31.7, and ASME Boiler and Pressure Vessel Code, Section IX, IS-400.

For normal opening, the integrity of the Reactor Coolant System, in terms of strength, is unchanged. If the system does not leak at 2285 psig (operating pressure +100 psi; $\pm$50 psi is normal system pressure fluctuation), it will be leak tight during normal operation (Reference 1).

REFERENCE

(1) UFSAR, Section 4.2.3.8 - "Leak Detection"

4.4.1.1.4 Conduct of Tests

a. During the period between the initiation of the containment inspection and the performance of a periodic integrated leakage rate test, no repairs or adjustments shall be made unless the inspection reveals structural deterioration which could affect the containment structural integrity or leak-tightness. Such structural deterioration shall be corrected before performance of the test and a description of the deterioration and the corrective action taken shall be reported as part of the test report submitted in accordance with Technical Specification 4.4.1.1.8.

b. The containment test pressure shall be allowed to stabilize for a period of not less than four hours prior to the start of a leakage rate test.

c. The test duration shall be at least 24 hours unless experience from at least two prior tests provides evidence of the adequacy of a shorter test duration.

d. Test accuracy shall be verified by supplementary means, such as measuring the quantity of air required to return to the starting point or by imposing a known leak rate to demonstrate the validity of measurements.

e. Closure of containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves without preliminary exercises or adjustment.

f. Portions of the following fluid systems will be drained and vented to containment atmosphere prior to and during the integrated leakage rate tests:

   1. Parts of the reactor coolant pressure boundary open directly to containment atmosphere under post accident conditions. (Become an extension of containment boundary)

   2. Portions of closed systems inside containment that penetrate containment and rupture as a result of a loss of coolant accident.

   NOTE: Systems that are required to maintain the plant in a safe condition during the tests and systems that are normally filled with water and operating under post-accident conditions need not be vented. In addition, missile shielded lines outside the secondary shield will not be vented.

g. All containment components normally pressurized by the penetration pressurization system shall be at atmospheric pressure during the integrated leakage rate tests.

detection tests. Sufficient data and analysis shall be included to show that a stabilized leak rate was attained and to identify all significant required correction factors such as those associated with humidity and barometric pressure, and all significant errors such as those associated with instrumentation sensitivities and data scatter. This report shall be titled "Reactor Containment Building Integrated Leak Rate Test" and shall be submitted to the NRC within 3 months of the test.

4.4.1.2 Local Leakage Rate Tests

4.4.1.2.1 Scope of Testing

Local Leakage Rate tests of penetrations and valves identified in the UFSAR shall be performed in accordance with 10 CFR 50 Appendix J except as provided in 4.4.1.2.5.f (Reference 1).

4.4.1.2.2 Conduct of Tests

a. Local leak rate tests shall be performed pneumatically at a pressure of not less than P_a, with the following exception: The access hatch door seal test shall normally be performed at 10 psig and the test every six months specified in 4.4.1.2.5.b shall be performed at a pressure not less than P_a.

b. Acceptable methods of testing are halogen gas detection, pressure decay, pneumatic flow measurement, or equivalent.

c. The pressure for a valve test shall be applied in the same direction as that when the valve would be required to perform its safety function unless it can be determined that the direction will provide equivalent or more conservative results.

d. Valves to be tested shall be closed by normal operation and without any preliminary exercising or adjustments.

4.4.1.2.3 Acceptance Criteria

The combined leakage from all penetrations and valves subject to Local Leak Rate tests shall not exceed .6 L_a (the maximum allowable leakage rate at P_a).

4.4.1.2.4 Corrective Action and Retest

a. If at any time it is determined that the criterion of 4.4.1.2.3 above is exceeded, repairs shall be initiated immediately.

b. If conformance to the criterion of 4.4.1.2.3 is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shutdown and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

Bases (1)

The reactor building is designed for an internal pressure of 55 psig and a steam-air mixture temperature of 281°F. Prior to initial operation, the containment was strength tested at 115 percent of design pressure and leak rate tested at the design pressure. The containment was also leak tested prior to initial operation at approximately 50 percent of the design pressure. These tests established the acceptance criteria of 4.4.1.1.3.

The performance of periodic integrated and local leakage rate tests during the plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions "as found" local leakage results must be documented for correction of the integrated leakage rate test results. Containment isolation valves are to be closed in the normal manner prior to local or integrated leakage rate tests. Containment Isolation Valves are addressed in the UFSAR (Reference 2).

The minimum test pressure of 30 psig for the periodic integrated leakage rate test is sufficiently high to provide an accurate measurement of the leakage rate and it exceeds the pre-operational leakage rate test at the reduced pressure of 27.5 psig. The specification provides a relationship for relating the measured leakage of air at the reduced pressure to the potential leakage of 55 psig. The minimum of 24 hours was specified for the integrated leakage rate test to help stabilize conditions and thus improve accuracy and to better evaluate data scatter. The frequency of the periodic integated leakage rate test is keyed to the refueling schedule for the reactor, because these tests can best be performed during refueling shutdowns.

The specified frequency of periodic integrated leakage rate tests is based on three major considerations. First is the low probability of leaks in the liner, because of conformance of the complete containment to a 0.10 percent leakage rate at 55 psig during pre-operational testing and the absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at design pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation and the low value of leakage that is specified as acceptable from penetrations and isolation valves, 0.6 L_a.

More frequent testing of various penetrations is specified as these locations are more susceptible to leakage than the reactor building liner due to the mechanical closure involved. The basis for specifying a total leakage rate of 0.6 L_a from those penetrations and isolation valves is that more than one-half of the allowable integrated leakage rate will be from these sources.

Valve operability tests are specified to assure proper closure or opening of the reactor building isolation valves to provide for isolation or functioning of Engineered Safety Features systems. Valves will be stroked to the position required to fulfill their safety function unless it is establish that such testing is not practical during operation. Valves that cannot be full-stroke tested will be part-stroke tested during operation and full-stroke tested during each normal refueling shutdown.

Periodic surveillance of the airlock interlock systems is specified to assure continued operability and preclude instances where one or both doors are inadvertently left open. When an airlock is inoperable and containment integrity is required, local supervision of airlock operation is specified.

Purge valve interspace pressurization test operability requirements and inspections provide a high degree of assurance of purge valve performance as containment isolation barriers.

<u>References</u>

(1) UFSAR, Chapter 5.7.4 - "Post Operational Leakage Rate Tests"

(2) UFSAR, Tables 5.7-1 and 5.7-3

4.4.2 Structural Integrity

Specification

4.4.2.1 Inservice Tendon Surveillance Requirements

The surveillance program for structural integrity and corrosion protection conforms to the recommendations of the U.S. NRC Regulatory Guide 1.35, proposed Revision 3, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures." The detailed surveillance program for the prestressing system tendons shall be based on periodic inspection and mechanical tests to be performed on selected tendons, as specified hereafter.

4.4.2.1.1 Containment Tendons

Tendon surveillance was completed for one, three and five years following initial structural integrity using a Tech. Spec. based on Regulatory Guide 1.35 Rev. 1. The containment tendon structural integrity shall be demonstrated at five year intervals thereafter by:

a. Determining that for a representative sample* of at least 23 tendons (6 dome, 7 vertical, and 10 hoop) each tendon has a lift off force equalling, or exceeding, its lower limit predicted for the time of the test as defined in NRC Regulatory Guide 1.35, "Inservice Inspection for Ungrouted Tendons in Prestressed Concrete Containments", Proposed Revision 3, April, 1979.

   If the lift off force of a selected tendon in a group lies between the prescribed lower limit and 90% of that limit, one tendon on each side of this tendon shall be checked for their lift off forces. If the lift off forces of the adjacent tendons are equal to, or greater than, their prescribed lower limits at the time of the test, the single deficiency shall be considered unique and acceptable.

   If the lift off force of any one tendon lies below 90% of its prescribed lower limit, the tendon shall be considered a defective tendon. It shall be completely detensioned and a determination made as to the cause of the occurrence.

   If the inspections performed at one, three, and five years indicate no abnormal degradation of the post-tensioning system, the number of tendons checked for lift off force during subsequent tests may be reduced to a representative sample of at least 11 tendons (3 dome, 3 vertical, and 5 hoop).

*For each inspection, the tendons shall be selected on a random but representative basis so that the sample group will change somewhat for each inspection; however, to develop a history of tendon performance and to correlate the observed data, one tendon from each group (dome, vertical, and hoop) may be kept unchanged after the initial selection (Reference 1).

b. Determining that the average of the normalized* tendon lift off forces for each tendon group (vertical, dome, and hoop) is equal to, of greater than 1010 Kips for vertical tendons, 1040 Kips for dome tendons, and 1121 Kips for hoop tendons. If this requirement is not met, an additional sample of 4%, with a minimum of four and a maximum of ten, of the same group of tendons shall be inspected. If the total population of each group of the sampled tendons meets the criteria above, the structural integrity of the containment shall be considered acceptable.

c. Detensioning one tendon in each group (dome, vertical and hoop) from the representative sample. One wire shall be removed from each detensioned tendon and examined to determine:

1. That over the entire length of the wire, the tendon wires have not undergone corrosion, cracks, or damage beyond that which was originally recorded and the extent of corrosion is within specified acceptable limits.
2. A minimum tensile strength value of 240,000 psi (guaranteed ultimate strength of the tendon material) for at least three wire samples (one from each end and one at mid-length) cut from each removed wire.

Upon retensioning, the elongation shall be within plus or minus 5% of that recorded at original stressing of the tendon. If the 5% limit is not met, an investigation shall be made to determine if wire failure is the cause.

d. Determining for each tendon in the above representative sample, that the sheathing filler grease is within acceptable limits as to:

1. presence of voids.
2. presence of free water.
3. chemical and physical properties.

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* In order for the tendon lift off forces to be indicative of the average level of prestress, each lift off force is adjusted for differences which exist among the tendons due to initial lock off force and elastic shortening loss.

References

(1) UFSAR, Section 5.7.5 - "Tendon Stress Surveillance"

Bases

The Emergency loading sequence and automatic power transfer controls the operation of the pumps associated with the emergency core cooling system and Reactor Building cooling system.

Automatic start and loading of the emergency diesel generator to meet the requirements of 4.5.1.1b/c above is described in Technical Specification 4.6.1.b.

c. When the Decay Heat System is required to be operable, the correct position of DH-V-19A/B shall be verified by observation within four hours of each valve stroking operation or valve maintenance, which effects the position indicator.

4.5.2.3 Core Flooding

a. During each refueling period, a system test shall be conducted to demonstrate proper operation of the system. During depressurization of the Reactor Coolant System, verification shall be made that the check and isolation valves in the core cooling flooding tank discharge lines operate properly.

b. The test will be considered satisfactory if control board indication of core flooding tank level verifies that all valves have opened.

4.5.2.4 Component Tests

a. At intervals not to exceed 3 months, the components required for emergency core cooling will be tested.

b. The test will be considered satisfactory if the pumps and fans have been successfully started and the valves have completed their travel as evidenced by the control board component operating lights, and either the station computer or pressure/flow indication.

Bases

The emergency core cooling systems (Reference 1) are the principal reactor safety features in the event of a loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The low pressure injection pumps are tested singularly for operability by opening the borated water storage tank outlet valves and the bypass valves in the borated water storage tank fill line. This allows water to be pumped from the borated water storgage tank through each of the injection lines and back to the tank.

The minimum acceptable HPI/LPI flow assures proper flow and flow split between injection legs.

With the reactor shutdown, the valves in each core flooding lines are checked for operability by reducing the reactor coolant system pressure until the indicated level in the core flood tanks verify that the check and isolation valves have opened.

References

(1) UFSAR, Section 6.1 - "Emergency Core Cooling System"

b. Reactor Building Cooling and Isolation Systems

1. During each refueling period, a system test shall be conducted to demonstrate proper operation of the system. A test signal will actuate the R.B. emergency cooling system valves to demonstrate operability of the coolers.

2. The test will be considered satisfactory if the valves have completed their expected travel as evidenced by the control board component operating lights, and either the station computer or local verification.

4.5.3.2 Component Tests

a. At intervals not to exceed three months, the components required for reactor building cooling and isolation will be tested.

b. The test will be considered satisfactory if the valves have completed their expected travel as evidenced by the control board component operating lights, and either the station computer or local verification.

Bases

The reactor building cooling and isolation systems and reactor building spray system are designed to remove the heat in the containment atmosphere to prevent the building pressure from exceeding the design pressure (References 1 and 2).

The delivery capability of one reactor building spray pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump.

With the pumps shut down and the borated water storage tank outlet closed, the reactor building spray injection valves can each be opened and closed by the operator action. With the reactor building spray inlet valves closed, low pressure air can be blown through the test connections of the reactor building spray nozzles to demonstrate that the flow paths are open.

The equipment, piping, valves and instrumentation of the reactor building cooling system are arranged so that they can be visually inspected. The cooling units and associated piping are located outside the secondary concrete shield. Personnel can enter the reactor building during power operations to inspect and maintain this equipment.

The reactor building fans are normally operating periodically, constituting the test that these fans are operable.

Reference

(1) UFSAR, Section 6.2 - "Reactor Building Spray System"

(2) UFSAR, Section 6.3 - "Reactor Building Emergency Cooling System"

4.5.4 DECAY HEAT REMOVAL SYSTEM LEAKAGE

Applicability
Applies to Decay Heat Removal System leakage.

Objective
To maintain a low leakage rate from the Decay Heat Removal System to prevent significant off-site exposures.

Specification

4.5.4.1 The maximum allowable leakage from the Decay Heat Removal System components as measured during refueling tests in Specification 4.5.4.2 shall not exceed six gallons per hour.

4.5.4.2 During each refueling period the following tests of the Decay Heat Removal System shall be conducted to determine leakage:

a. The portion of the Decay Heat Removal System, except as specified in "b", that is outside containment shall be leak tested either by use in normal operation or by hydrostatically testing at 350 psig.

b. Piping from the Reactor Building Sump to the Decay Heat Removal System pump suction isolation valve shall be pressure tested at no less than 55 psig.

c. Visual inspection shall be made for leakage from components of the system. Leakage shall be measured by collection and weighing or by another equivalent method.

Bases

The leakage rate limit for the Decay Heat Removal System is a judgement value based on ensuring that its components can be expected to operate for an extended period (200 days or more) after a loss-of-coolant accident without significant leakage (Reference 1). The test pressure achieved either by normal system operation or by hydro- static testing (350 psig) provides an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the test pressure for the recirculation lines from the reactor building sump to the decay heat system (55 psig) is the design pressure of the reactor building. The dose to the thyroid calculated as a result of the acceptance limit leakage rate (4.5.4.1) is 0.39 rem for a 2 hour exposure at the site boundary (Reference 2).

REFERENCE

(1) UFSAR, Section 6.4.4 - "Design Basis Leakage" and Table 6.4-3 - "Leakage Quantities to the Auxiliary Building"

(2) UFSAR, Section 14.2.2.5(d) - "Effects of Engineered Safeguards Leakage During Maximum Hypothetical Accident"

4.6 EMERGENCY POWER SYSTEM PERIODIC TESTS

Applicability: Applies to periodic testing and surveillance requirement of the emergency power system.

Objective: To verify that the emergency power system will respond promptly and properly when required.

Specification:

The following tests and surveillance shall be performed as stated:

4.6.1 Diesel Generators

a. Manually-initiate start of the diesel generator, followed by manual synchronization with other power sources and assumption of load by the diesel generator up to the name-plate rating (3000 kw). This test will be conducted every month on each diesel generator. Normal plant operation will not be effected.

b. Automatically start and loading the emergency diesel generator in accordance with Specification 4.5.1.1.b/c including the following. This test will be conducted every refueling interval on each diesel generator.

   (1) Verify that the diesel generator starts from ambient condition upon receipt of the ES signal and is ready to load in $\leq$10 seconds.

   (2) Verify that the diesel block loads upon simulated loss of offsite power in $\leq$30 seconds.

   (3) The diesel operates with the permanently connected and auto connected load for $\geq$5 minutes.

   (4) The diesel engine does not trip when the generator breaker is opened while carrying emergency loads.

   (5) The diesel generator block loads and operates for $\geq$5 minutes upon reclosure of the diesel generator breaker.

c. Each diesel generator shall be given an inspection at least annually in accordance with the manufacturer's recommendations for this class of stand-by service.

4.6.2 Station Batteries

a. The voltage, specific gravity, and liquid level of each cell will be measured and recorded monthly.

b. The voltage and specific gravity of a pilot cell will be measured and recorded weekly.

c. Each time data is recorded, new data shall be compared with old to detect signs of abuse or deterioration.

d. The battery will be subjected to a load test at a frequency not to exceed refueling periods. The battery voltage as a function of time will be monitored to establish that the battery performs as expected during this load test.

4.6.3 Pressurizer Heaters

a. The following tests shall be conducted at least once each refueling:

(1) Pressurizer heater groups 8 and 9 shall be transferred from the normal power bus to the
Upon completion of this test, the heaters shall be
returned emergency power bus and energized.
to their normal power bus.

(2) Demonstrate that the pressurizer heaters breaker on the emergency bus cannot be closed until the safeguards signal is bypassed and can be closed following bypass.

(3) Verfiy that following input of the Engineered Safeguards Signal, the circuit breakers, supplying power to the manually transferred loads for pressurizer heater groups 8 and 9, have been tripped.

Bases

The tests specified are designed to demonstrate that one diesel generator will provide power for operation of safeguards equipment. They also assure that the emergency generator control system and the control systems for the safeguards equipment will function automatically in the event of a loss of normal a-c station service power or upon the receipt of an engineered safeguards Actuation Signal. The automatic tripping of manually transferred loads, on an Engineered Safeguards Actuation Signal, protects the diesel generators from a potential over-load condition. The testing frequency specified is intended to identify and permit correction of any mechanical or electrical deficiency before it can result in a system failure. The fuel oil supply, starting circuits, and controls are continuously monitored and any faults are alarmed and indicated. An abnormal condition is these systems would be signaled without having to place the diesel generators on test.

Precipitous failure of the station battery is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails.

The PORV has a remotely operated block valve to provide a positive shutoff capability should the relief valve become inoperable. The electrical power for both the relief valve and the block valves is supplied from an ESF power source to ensure the ability to seal this possible RCS leakage path.

The requirement that a minimum of 107 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation.

4.7 REACTOR CONTROL ROD SYSTEM TESTS

4.7.1 CONTROL ROD DRIVE SYSTEM FUNCTIONAL TESTS

Applicability

Applies to the surveillance of the control rod system.

Objective

To assure operability of the control rod system.

Specification

4.7.1.1 The control rod trip insertion time shall be measured for each control rod at either full flow or no flow conditions following each refueling outage prior to return to power. The maximum control rod trip insertion time for an operable control rod drive mechanism, except for the axial power shapping rods (APSRs), from the fully withdrawn position to 3/4 insertion (104 inches travel) shall not exceed 1.66 seconds at hot reactor coolant full flow conditions or 1.40 seconds for the hot no flow conditions (Reference 1). For the APSRs it shall be demonstrated that loss of power will not cause rod movement. If the trip insertion time above is not met, the rod shall be declared inoperable.

4.7.1.2 If a control rod is misaligned with its group average by more than an indicated nine inches, the rod shall be declared inoperable and the limits of Specification 3.5.2.2 shall apply. The rod with the greatest misalignment shall be evaluated first. The position of a rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operablility of rods with lesser misalignments.

4.7.1.3 If a control rod cannot be exercised, or if it cannot be located with absolute or relative position indications or in or out limit lights, the rod shall be declared to be inoperable.

Bases

The control rod trip insertion time is the total elapsed time from power interuption at the control rod drive breakers until the control rod has actuated the 25% withdrawn reference switch during insertion from the fully withdrawn position. The specificed trip time is based upon the safety analysis in UFSAR, Chapter 14 and the Accident Parameters as specified therein.

Each control rod drive mechanism shall be exercised by a movement of approximately two inches of travel every two weeks. This requirement shall apply to either a partial or fully withdrawn control rod at reactor operating conditions. Exercising the drive mechanisms in this manner provides assurance of reliability of the mechanisms.

A rod is considered inoperable if it cannot be exercised, if the trip insertion time is greater than the specified allowable time, or if the rod deviates from its group average position by more than nine inches. Conditions for operation with an inoperable rod are specified in Technical Specification 3.5.2.

REFERENCE

(1) UFSAR, Section 3.1.2.4.3 - "Control Rod Drive Mechanism"

4.7.2 CONTROL ROD PROGRAM VERIFICATION (Group vs. Core Positions)

Applicability

Applies to surveillance of the control rod systems.

Objective

To verify that the designated control rod (by core position 1 through 69) is operating in its programmed functional position and group. (rod 1 through 12, group 1-8)

Specification

4.7.2.1 Whenever the control rod drive patch panel is locked (after inspection, test, reprogramming, or maintenance) each control rod drive mechanism shall be selected from the control room and exercised by a movement of two inches or less to verify that the proper rod has responded as shown on the unit computer printout of the rod or on the input to the computer for that rod.

4.7.2.2 Whenever power or instrumentation cables to the control rod drive assemblies atop the reactor or at the bulkhead are disconnected or removed, an independent verification check of their reconnection shall be performed.

4.7.2.3 Any rod found improperly programmed shall be declared inoperable until properly programmed.

Bases

Each control rod has a relative and an absolute position indicator system. One set of outputs goes to the plant computer identified by a unique core coordinate associated with only one core position. The other set of outputs goes to a programmable bank of 69 edgewise meters in the control room. In the event that a patching error is made in the patch panel or connectors in the cables leading to the control rod drive assemblies or to the control room meter bank are improperly transposed upon reconnection, these errors and transpositions will be discovered by a comparative check by (1) selecting a specific rod from one group (e.g., rod 1 in regulating group 6) (2) noting that the program-approved core position for this rod of the group (assume the approved core position is 8-D) (3) exercise the selected rod and (4) note that (a) the computer displays both absolute and relative position response for the approved core position (assumed to be position 8-D) (b) the proper meter in the control room display bank (assumed to be rod 1 in group 6) in both absolute and relative meter positions. This type of comparative check will not assure detection of improperly connected cables inside the Reactor Building. For these, (Section 4.7.2.2) it will be necessary for a responsible person, other than the one doing the work, to verify by appropriate means that each cable has been matched to the proper control rod drive assembly.

4.8 MAIN STEAM ISOLATION VALVES

Applicability

Applies to the periodic testing of the main steam isolation valves.

Objective

To specify the minimum frequency and type of tests to be applied to the main steam isolation valves.

Specification

4.8.1 A check of valves stem movement, up to 10 percent, shall be performed on a monthly basis when the unit is operational and under normal flow and load conditions.

4.8.2 The main steam isolation valves shall be tested at intervals not to exceed the normal refueling outage. Closure time of <120 seconds shall be verified. This test will be performed under no flow and no load conditions.

Bases

Since a portion of the main steam lines and the steam lines to the main feed pump turbines are located in the turbine hall which is not protected against hypothetical tornado, missile, or aircraft incident; main steam isolation stop check valves are provided and located in the hardened portion of the intermediate building. These stop check valves are remotely closed by the operator from the control room, close in less than two minutes, and are tight closing for long term containment isolation (Reference 1). Their ability to close upon signal should be verified at intervals not to exceed each scheduled refueling shutdown, and valve stem freedom should be checked on a monthly basis.

References

(1) UFSAR, Section 10.3.1 - "Main Steam System" and Table 10.3-1 - "Main Steam Component Data"

4.12.2 REACTOR BUILDING PURGE AIR TREATMENT SYSTEM

Applicability: Applies to the reactor building purge air treatment system and associated components (Reference 1).

Objective: To verify that this system and associated components will be able to perform its design functions.

Specification

4.12.2.1 At least once per refueling interval or once per 18 months, whichever comes first it shall be demonstrated that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at system design flow rate ($\pm$10%).

4.12.2.2 a. The tests and sample analysis required by Specification 3.15.2.2, shall be performed initially, once per refueling interval or 2 years, whichever comes first, or within 30 days prior to the movement of irradiated fuel in containment and following significant painting, steam, fire, or chemical release in any ventilation zone communicating with the system that could contaminate the HEPA filters or charcoal adsorbers.

b. DOP testing shall be performed after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing which could affect HEPA frame bypass leakage.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing which could affect the charcoal adsorber bank bypass leakage.

d. The DOP and halogenated hydrocarbon testing shall be performed at the maximum available flow considering physical restrictions, i.e., purge valve position, and gaseous radioactive release criteria.

e. Each refueling, AH-E7A&B shall be shown to operate within $\pm$ 5000 cfm of design flow (50,000 cfm) with purge valves fully open.

4.12.2.3 An air distribution test shall be performed on the HEPA filter bank initially and after any maintenance or testing that could affect the air distribution within the system. The air distribution across the HEPA filter bank shall be uniform within $\pm$20%. The test shall be performed at 50,000 cfm ($\pm$10%) flow rate with purge valves fully open.

Bases

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 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 every refueling interval to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with approved test procedures. 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 should be replaced with an adsorbent qualified according to Regulatory Guide 1.52, March 1978. Tests of the HEPA filters with DOP aerosol shall also be performed in accordance with approved test procedures. Any HEPA filters found defective should be replaced with filters qualified according to Regulatory Guide 1.52, March 1978.

Fans AH-E7A&B performance verification is necessary to ensure adequate flow to perform the filter surveillance of T.S. 4.12.2.1 and 4.12.2.3 and can only be demonstrated by running both fans simultaneously. This can only be accomplished when purge valves are note limited to 30° open (i.e., cold shutdown).

Since H_2 purge has been superseded by the installation of H_2 recombiners at TMI-I, the reactor building purge exhaust system no longer is relied upon to serve an operating accident mitigating (i.e. LOCA) function. The retest requirement of T.S. 4.12.2.2a has therefore been changed to reflect the same retest requirements as the auxiliary and fuel handling building ventilation system which similarly serves no operating accident mitigating function.

If significant painting, steam, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the Director Operations and Maintenance - TMI-1.

References

(1) UFSAR, Section 5.6 - "Ventilation and Purge Systems"

Bases

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 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 every refueling interval to show system performance capability.

Tests and sample analysis assure that 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. The in-place test criteria and laboratory test criteria for activated charcoal will meet the guidelines of ANSI-N510-1980. If test results are unacceptable, all adsorbent in the system should be replaced with an adsorbent qualified according to Regulatory Guide 1.52, March 1978 or ANSI-N509-1980. Any HEPA filters found defective should be replaced with filters qualified according to Regulatory Guide 1.52, March 1978 or ANSI-N509-1980.

If significant painting, steam, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of what is significant shall be made by the Director Operations & Maintenance - TMI-1.

Operation of the Auxiliary and Fuel Handling Building Exhaust Fans each month for at least ten (10) hours will demonstrate operability of the fans.

Bases

The FHB ESF Air Treatment System is a system which is normally kept in a "standby" operating status. Tests and sample analysis assure that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedure should allow for the removal of a sample from one adsorber test cannister. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. The in-place test criteria and laboratory test criteria for activated charcoal will meet the guidelines of ANSI-N510-1980. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified in accordance with ANSI-N509-1980. Any HEPA filters found defective will be replaced with filters qualified in accordance with ANSI-N509-1980.

Pressure drop across the entire filtration unit of less than 7.0 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.

Operation of the system for 10 hours every month will demonstrate operability of the filters and adsorber system and remove excessive moisture buildup on the adsorbers and HEPA filters.

If significant painting, steam, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational movement of irradiated fuel. The determination of what is significant shall be made by the Director Operations & Maintenance - TMI-1.

4.19.4 Acceptance Criteria (Continued)

6. Repair Limit means the extent of degradation at or beyond which the tube shall be repaired or removed from service because it may become unserviceable prior to the next inspection.

This limit is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss of coolant accident, or a steam line or feedwater line break as specified in 4.19.3.c., above.

8. Tube Inspection means an inspection of the steam generator tube from the bottom of the upper tubesheet completely to the top of the lower tubesheet, except as permitted by 4.19.2.b.2, above.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (removal from service by plugging, or repair by kinetic expansion, sleeving, or other methods, of all tubes exceeding the repair limit and all tubes containing throughwall cracks) required by Table 4.19-2.

4.19.5 Reports

a. Following the completion of each inservice inspection of steam generator tubes, the number of tubes repaired or removed from service in each steam generator shall be reported to the NRC within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported to the NRC within 12 months following completion of the inspection.

This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes repaired or removed from service.

c. Results of steam generator tube inspections which fall into Category C-3 require notification in accordance with 10 CFR 50.72 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence in accordance with 10 CFR 50.73.

5.2 CONTAINMENT

Applicability

Applies to those design features of the containment system relating to operational and public safety.

Objective

To define the significant design features of the reactor containment.

Specification

Containment consists of two systems which are the reactor building and reactor building isolation system.

5.2.1 REACTOR BUILDING

The reactor building completely encloses the reactor and the associated reactor coolant systems. The reactor building is a reinforced concrete structure composed of cylindrical walls with a flat foundation mat, and a shallow dome roof. The foundation slab is reinforced with conventional mild-steel reinforcing. The cylindrical walls are prestressed with a post-tensioning tendon system in the vertical and horizontal directions. The dome roof is prestressed utilizing a three-way post-tensioning tendon system. The inside surface of the reactor building is lined with a carbon steel liner to ensure a high degree of leak tightness for containment.

The internal free volume of the reactor building is in excess of 2.0×10^6 cubic feet. The foundation mat is 9 ft thick with a 2 ft thick concrete slab above the bottom liner plate. The cylindrical portion has an inside diameter of 130 ft, wall thickness of 3 ft 6 in., and a height of 157 ft from top of foundation slab to the spring line. The shallow dome roof has a large radius of 110 ft, a transition radius of 20 ft 6 in., a thickness of 3 ft, and an overall height of 32 ft 4 1/8 in.

The concrete containment building provides adequate biological shielding for both normal operation and accident situations. Design pressure and temperature are 55 psig and 281°F, respectively. The reactor building is designed for an external atmospheric pressure of 2.5 psi greater than the internal pressure.

Penetration assemblies are welded to the reactor building liner. Access openings, electrical penetrations, and fuel transfer tube covers are equipped with double seals. Reactor building purge penetrations and reactor building atmosphere sampling penetrations are equipped with double valves having resilient seating surfaces (Reference 1).

The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a loss of coolant accident, as defined in Section 14, with no loss of integrity. In this event the total energy contained in the water of the reactor coolant system is assumed to be released into the reactor building through a break in the reactor coolant piping. Subsequent pressure behavior is determined by the building volume, engineered safeguards, and the combined influence of energy sources and heat sinks.

5.2.2 REACTOR BUILDING ISOLATION SYSTEM

Leakage through all fluid penetrations not serving accident-consequence-limiting systems is minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building and various types of isolation valves (Reference 2).

REFERENCES

(1) UFSAR Section 5.2.2.4.8 - "Penetrations and Openings"

(2) UFSAR Section 5.3.1 - "Isolation System - Design Bases"

5.3 REACTOR

Applicability

Applies to the design features of the reactor core and reactor coolant system.

Objective

To define the significant design features of the reactor core and reactor coolant system.

Specification

5.3.1 REACTOR CORE

5.3.1.1 The reactor core is composed of slightly enriched uranium dioxide pellets contained in fuel rods. A fuel assembly normally contains 208 fuel rods arranged in a 15 by 15 lattice. The details of the fuel assembly design are described in TMI-1 UFSAR Chapter 3.

5.3.1.2 The reactor core shall approximate a right circular cylinder with an equivalent diameter of 128.9 inches. The active fuel height is defined in TMI-1 UFSAR Chapter 3.

5.3.1.3 The core average and individual batch enrichments for the present cycle are described in TMI-1 UFSAR Chapter 3.

5.3.1.4 The control rod assemblies (CRA) and axial power shaping rod assemblies (APSRA) are distributed in the reactor core as shown in TMI-1 FSAR Chapter 3. The CRA and APSRA design data are also described in the UFSAR.

5.3.1.5 The TMI-1 core may contain burnable poison rod assemblies (BPRA) as described in TMI-1 UFSAR Chapter 3.

5.3.1.6 Reload fuel assemblies and rods shall conform to design and evaluation data described in the UFSAR and shall not exceed an enrichment of 4.3 weight percent of U^{235}.

5.3.2 REACTOR COOLANT SYSTEM

5.3.2.1 The reactor coolant system shall be designed and constructed in accordance with code requirements. (Refer to UFSAR Chapter 4 for details of design and operation.)

5.3.2.2 The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, shall be designed for a pressure of 2,500 psig and a temperature of 650 F. The pressurizer and pressurizer surge line shall be designed for a temperature of 670 F.

5.3.2.3 The reactor coolant system volume shall be less than 12,200 cubic feet.

5.4.2 SPENT FUEL STORAGE (Reference 1)

a. Irradiated fuel assemblies will be stored, prior to offsite shipment, in the stainless steel lined spent fuel pools, which are located in the fuel handling building.

b. Whenever there is fuel in the pool except for initial fuel loading, the spent fuel pool is filled with water borated to the concentration used in the reactor cavity and fuel transfer canal.

c. Spent fuel may also be stored in storage racks in the fuel transfer canal when the canal is at refueling level.

d. The fuel assembly storage racks provided and the number of fuel elements each will store are listed by location below:

	South End of Fuel Transfer Canal RB	Spent Fuel Pool A North End of Fuel Handling Building	Spent Fuel Pool B South End of Fuel Handling Building	Dry New Fuel Storage Area Fuel Handling Building
Fuel Assys	64 *	256 **	496 ***	66****
Cores	0.36	1.45	2.8	0.37

NOTES:
* Includes one space for accommodating a failed fuel detection container.
** Includes three spaces for accommodating failed fuel containers.
*** Spent Fuel Pool B contains spent fuel storage racks with a reduced center-to-center spacing of 13 5/8 inches to increase the storage capacity of the pool.
**** Includes twelve spaces which are required to be vacant of fissile or moderating material so that there is sufficient neutron leakage.

e. All of the fuel assembly storage racks provided are designed to Seismic Class 1 criteria to the accelerations indicated below:

	Fuel Transfer Canal in Reactor Building	Fuel Handling Building Dry New Fuel Storage Area And Spent Fuel Pool A	Fuel Handling Building Spent Fuel Pool B
Horiz.	0.76 g	0.38 g	*
Vertical	0.51 g	0.25 g	*

* The "B" pool fuel storage racks are designed using the floor response spectra of the Fuel Handling Building.

f. Fuel in the storage pool shall have a U-235 loading equal to or less than 57.8 grams of U-235 per axial centimeter of fuel assembly.

REFERENCES

(1) UFSAR, Section 9.7 - "Fuel Handling System"

ACTIVITIES

6.5.1.1 Each procedure required by Technical Specification 6.8 and other procedures which affect nuclear safety, and substantive changes thereto, shall be prepared by a designated individual(s)/group knowledgeable in the area affected by the procedure. Each such procedure, and substantive changes thereto, shall be reviewed for adequacy by an individual(s)/group other than the preparer, but who may be from the same organization as the individual who prepared the procedure or change.

6.5.1.2 Proposed changes to the Appendix "A" Technical Specifications shall be reviewed by a knowledgeable individual(s)/group other than the individual(s) group who prepared the change.

6.5.1.3 Proposed modifications that affect nuclear safety to unit structures, systems and components shall be designed by an individual/organization knowledgeable in the areas affected by the proposed modification. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification but may be from the same division as the individual who designed the modification.

6.5.1.4 Proposed tests and experiments that affect nuclear safety shall be reviewed by a knowledgeable individual(s)/group other than the preparer but who may be from the same division as the individual who prepared the tests and experiments.

6.5.1.5 Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, shall be reviewed by a knowledgeable individual(s)/group other than the individual/group which performed the investigation.

6.5.1.6 All REPORTABLE EVENTS shall be reviewed by an individual/group other than the individual/group which prepared the report.

6.5.1.7 Special reviews, investigations or analyses and reports thereon as requested by the Vice President TMI-1 shall be performed by a knowledgeable individual(s)/group.

6.5.1.8 The Security Plan and implementing procedures shall be reviewed by a knowledgeable individual(s)/group other than the individual(s)/group which prepared them.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Nuclear Regulatory Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR 50, and

b. Each REPORTABLE EVENT shall undergo an independent safety review pursuant to Specification 6.5.2.5.d.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a safety limit is violated:

a. The reactor shall be shutdown and operation shall not be resumed until authorized by the Nuclear Regulatory Commission.

b. An immediate report shall be made to the Director Operations and Maintenance, and Vice President TMI-1, and the event shall be reported to NRC in accordance with 10 CFR 50.72.

c. A complete analysis of the circumstances leading up to and resulting from the occurrence shall be prepared by the unit staff. This report shall include analysis of the effects of the occurrence and recommendations concerning operation of the unit and prevention of recurrence. This report shall be submitted to the Director Operations and Maintenance and the Vice President, TMI-1. The safety limit violation report shall be submitted to NRC in accordance with 10 CFR 50.73.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the items referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Surveillance and test activities of equipment that affects nuclear safety and radioactive waste management equipment.

c. Refueling Operations.

d. Security Plan Implementation.

e. Fire Protection Program Implementation.

f. Emergency Plan Implementation.

g. Process Control Program Implementation.

h. Offsite Dose Calculation Manual Implementation.

i. Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 4.15.

j. Plant Staff Overtime, to limit the amount worked by staff performing safety-related functions in accordance with NRC Policy Statement on working hours (Generic Letter No. 82-12).

6.8.2 Further, each procedure required by 6.8.1 above, and substantive changes thereto, shall be reviewed and approved as described in 6.5.1 prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of GPUNC Management Staff qualified in accordance with 6.5.1.14 and knowledgeable in the area affected by the procedure. For changes which may affect the operational status of unit systems or equipment, at least one of these individuals shall be a member of unit management or supervision holding a Senior Reactor Operator's License on the unit.

c. The change is documented, reviewed and approved as described in 6.5.1 within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Administrator of the NRC Region 1 Office unless otherwise noted.

6.9.1 Routine Reports

A. Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the UFSAR, Chapter 13 and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described.

Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

B. Annual Reports. Annual reports covering the activities of the unit as described below during the previous calendar year shall be submitted prior to March 1 of each year. (A single submittal may be made for the station. The submittal should combine those sections that are common to both units at the station.)

1. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions, (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions. (This tabulation supplements the requirements of Section 20.407 of 10 CFR Part 20.)