| 2.0 SAFETY LIMITS | LIMITING SAFETY SYSTEM SETTINGS |
|---|---|

$$S = \frac{486{,}000\ P}{X}$$

Where:

P = percent of rated power
X = peak heat flux - $(BTU/HR/FT^2)$
shall be used.

2. IRM--Flux Scram setting shall be ≤ 15% of rated neutron flux.

B. When the reactor pressure is less than 600 psig or core flow is less than 5% of design, the reactor thermal power transferred to the coolant shall not exceed 300 MW.

B. APRM Rod Block - The APRM rod block setting shall be as shown in Figure 2.3.1 unless the combination of power and peak heat flux is above the curve in Figure 2.3.2. When the combination of power and peak flux is above the curve in Figure 2.3.2, a rod block trip setting (RB) as given by:

$$RB = \frac{437{,}400P}{X}$$

where:

P = percent of rated power
X = peak heat flux $(BTU/HR/FT^2)$

shall be used.

C. 1. The neutron flux shall not exceed the scram setting established in Specification 2.3.A for longer than 0.95 seconds as indicated by the process computer.

C. Reactor Low Water Level Scram setting shall be ≥ 10'6" above the top of the active fuel.



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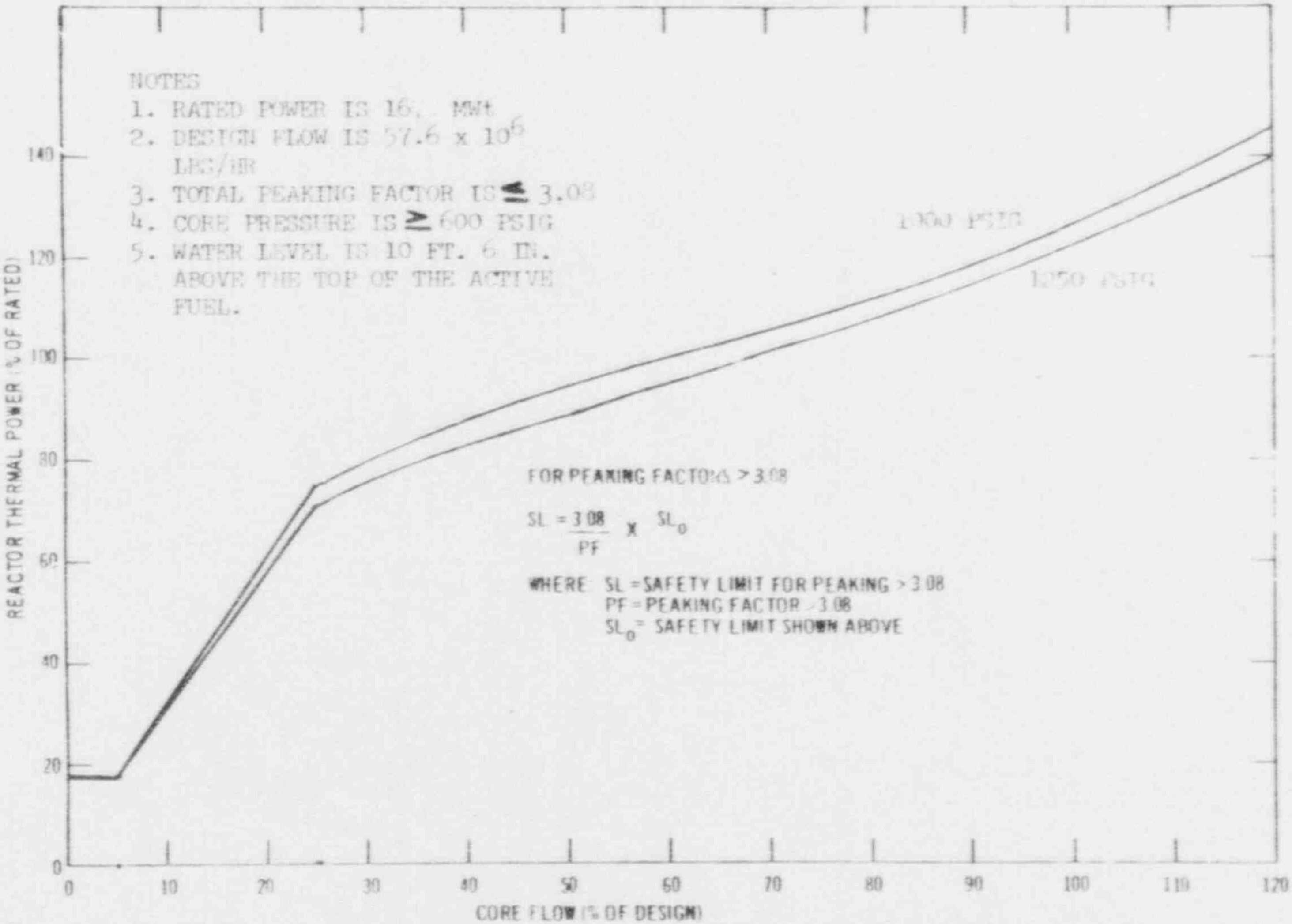


FIGURE 2.1.1 FUEL CLADDING INTEGRITY SAFETY LIMIT

Bases Continued:

The feedwater temperature assumed was the maximum design temperature output of the feedwater heaters at the given pressures and flows, which is 376°F for rated thermal power. For any lower feedwater temperature, sub-cooling is increased and the curves are conservative.

The water level assumed in the calculation of the safety limit was that level corresponding to the bottom of the steam separator skirt (7" on the level instrument is equivalent to 10'6" above the top of the active fuel at rated power). As long as the water level is above this point, the safety limit curves are applicable; i.e., the amount of steam carry under would not be increased, and, therefore, the core inlet enthalpy and sub-cooling would not be influenced.

The values of the parameters involved in Figure 2.1.1 can be determined from information available in the control room. Reactor pressure and flow are recorded and the Average Power Range Monitor (APRM) in-core nuclear instrumentation is calibrated to read in terms of percent power.

The range in pressure and flow used for Specification 2.1.A was 600 psig to 1250 psig and 5% to 100% flow respectively. Specification 2.1.B requires a restriction on power level when operating below 600 psig or 5% flow. In general, Specification 2.1.B will only be applicable during startup or shutdown of the plant. A review of all the applicable low pressure and low flow data (2, 3) has shown the lowest data point for transition boiling to have a heat flux of 144,000 BTU/HR/Ft$^2$. To assure applicability to Monticello fuel geometry and provide some margin, a factor of 1/2 was used to obtain the critical heat flux; i.e., critical heat flux was assumed to occur for these conditions at 72,000 BTU/HR/Ft$^2$. Assuming a peaking factor of 3.08, this is equivalent to a core average power of approximately 300 MW(t) (18% of rated). This value is applicable to ambient pressure and no flow conditions. For any greater pressure or flow conditions, there is increased margin.

---

(2) E. Janssen - "Multirod Burnout at Low Pressure" - ASME Paper 62-HT-26, August 1962.
(3) K. M. Becker - "Burnout Conditions for Flow of Boiling Water in Vertical Rod Clusters" - AE-74 (Stockholm, Sweden), May, 1962.

TABLE 3.1.1
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

| Trip Function | Limiting Trip Settings | Modes in which function must be Operable or Operating** Refuel(3) | Startup | Run | Total No. of Instrument Channels per Trip System | Min. No. of Operable or Operating Instrument Channels Per Trip System (1) | Required Condition |
|---|---|---|---|---|---|---|---|
| 1. Mode Switch in Shutdown | | x | x | x | 1 | 1 | A |
| 2. Manual Scram | | x | x | x | 1 | 1 | A |
| 3. Neutron Flux IRM (See Note 2) a. High-High b. Inoperative | ≤ 120/125 of full scale | x | x | x(c) | 4 | 3 | A |
| 4. Flow Referenced Neutron Flux APRM (See Note 5) a. High-High b. Inoperative c. Downscale | See Specifications 2.3A.1 ≤3/125 of full scale | | | x | 3 | 2 | A or B |
| 5. High Reactor Pressure | ≤ 1075 psig | x | x(f) | x(f) | 2 | 2 | A |
| 6. High Drywell Pressure | ≤ 2 psig | x(4) | x(e,f) | x(e,f) | 2 | 2 | A |
| 7. Reactor Low Water Level | ≥ 7 in.(6) | x | x(f) | x(f) | 2 | 2 | A |
| 8. Scram Discharge Volume High Level | ≤ 32 gal.(8) | x(a) | x(f) | x(f) | 2 | 2 | A |
| 9. Turbine Condenser Low Vacuum | ≥ 23 in. Hg | x(b) | x(b,f) | x(f) | 2 | 2 | A or C |

Table 3.2.1 - Continued

| Function | Trip Settings | Total No. of Instrument Channels Per Trip System | Min. No. of Operable or Operating Instrument Channels Per Trip System (1,2) | Required Conditions |
|---|---|---|---|---|
| b. High Drywell Pressure (5) | ≤ 2 psig | 2 | 2 | D |
| 3. Reactor Cleanup System (Group 3) | | | | |
| a. Low Reactor Water Level | ≥10'6" above the top of the active fuel | 2 | 2 | E |
| 4. HPCI Steam Lines | | | | |
| a. HPCI High Steam Flow | ≤150,000 lb/hr with ≤60 second time delay | 2(4) | 2 | F |
| b. HPCI High Steam Flow | ≤300,000 lb/hr | 2(4) | 2 | F |
| c. HPCI Steam Line Area High Temp. | ≤200°F | 16(4) | 16 | F |
| 5. RCIC Steam Lines | | | | |
| a. RCIC High Steam Flow | ≤45,000 lb/hr | 2(4) | 2 | G |
| b. RCIC Steam Line Area High Temp. | ≤200°F | 16(4) | 16 | G |

Table 4.2.1 - Continued
Minimum Test and Calibration Frequency For Core Cooling
Rod Block and Isolation Instrumentation

| Instrument Channel | Test (3) | Calibration (3) | Sensor Check (3) |
|---|---|---|---|
| 3. Steam Line Low Pressure | Note 1 | Once/3 months | None |
| 4. Steam Line High Radiation | Once/week (5) | Note 6 | Once/shift |
| HPCI ISOLATION | | | |
| 1. Steam Line High Flow | Note 1 | Once/3 months | None |
| 2. Steam Line High Temperature | Note 1 | Once/3 months | None |
| RCIC ISOLATION | | | |
| 1. Steam Line High Flow | Note 1 | Once/3 months | None |
| 2. Steam Line High Temperature | Note 1 | Once/3 months | None |
| REACTOR BUILDING VENTILATION | | | |
| 1. Radiation Monitors (Plenum) | Note 1 | Once/3 months | Once/shift |
| 2. Radiation Monitors (Refueling Floor) | Note 1 | Once/3 months | (4) |
| OFF GAS ISOLATION | | | |
| 1. Radiation Monitors | Notes (1,5) | Note 6 | Once/shift |

NOTES:

(1) Initially once per month until exposure hours (M as defined on Figures 4.1.1) is $2.0 \times 10^5$, thereafter according to Figure 4.1.1, with an interval not greater than three months.

Table 3.2.5 - Continued
Trip Function and Deviations

| | Trip Function | Deviation |
|---|---|---|
| Instrumentation That Initiates Emergency Core Cooling Systems Table 3.2.2 | Low-Low Reactor Water Level | -3 Inches |
| | Reactor Low Pressure (Pump Start) Permissive | -10 psi |
| | High Drywell Pressure | +1 psi |
| | Low Reactor Pressure (Valve Permissive | -10 psi |
| Instrumentation That Initiates Rod Block Table 3.2.3 | IRM Downscale | -2/125 of Scale |
| | IRM Upscale | +2/125 of Scale |
| | APRM Downscale | -2/125 of Scale |
| | APRM Upscale | See Basis 2.3 - Page 24 |
| | RBM Downscale | -2/125 of Scale |
| | RBM Upscale | Same as APRM Upscale |

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip settings, or, when a sufficient number of devices have been affected by any means such that the automatic function is incapable of operating within the allowable deviation while in a reactor mode in which the specified function must be operable or when actions specified are not initiated as specified.

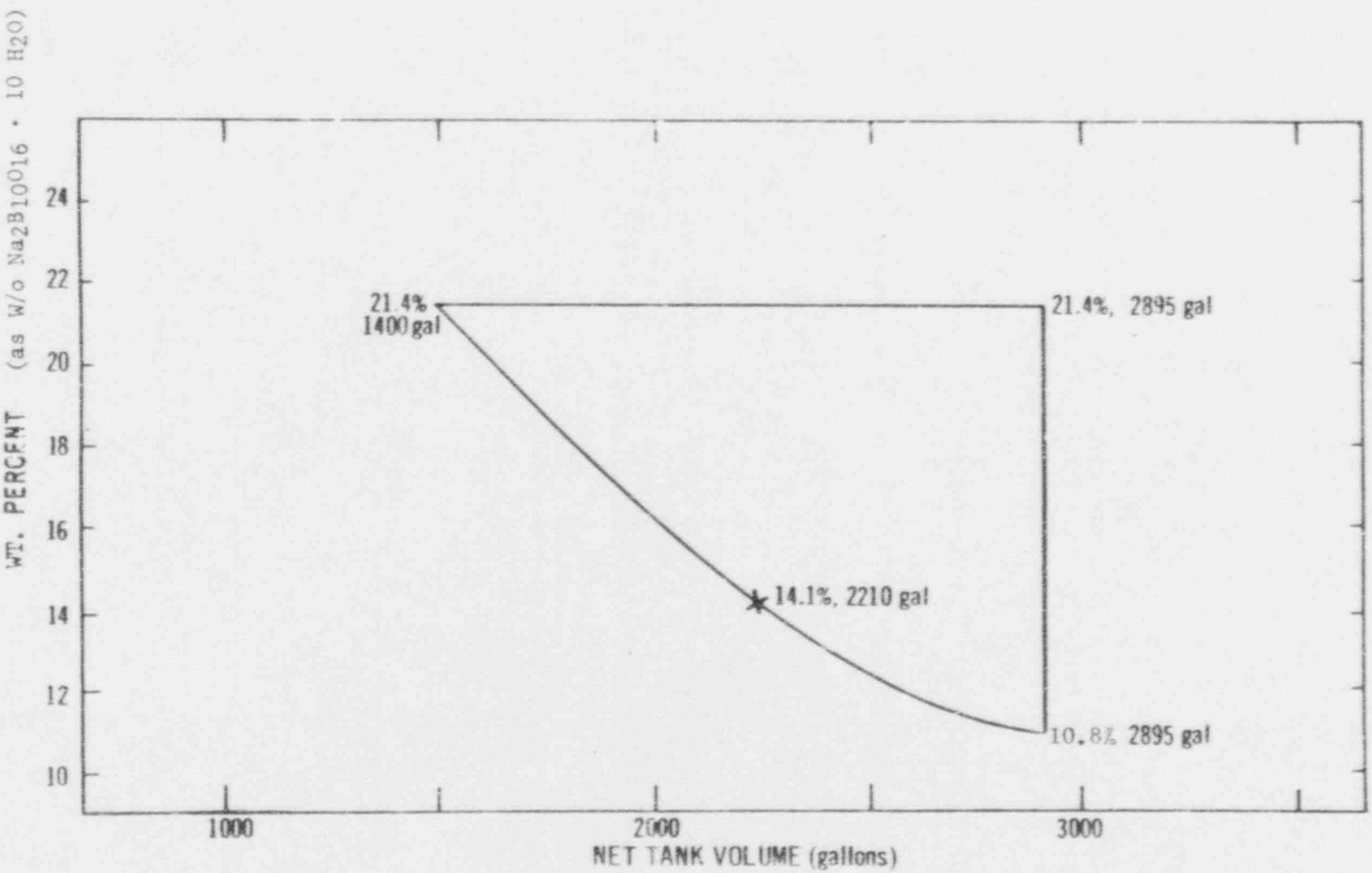


FIGURE 3.4.1. Sodium Pentaborate Solution Volume — Concentration Requirements

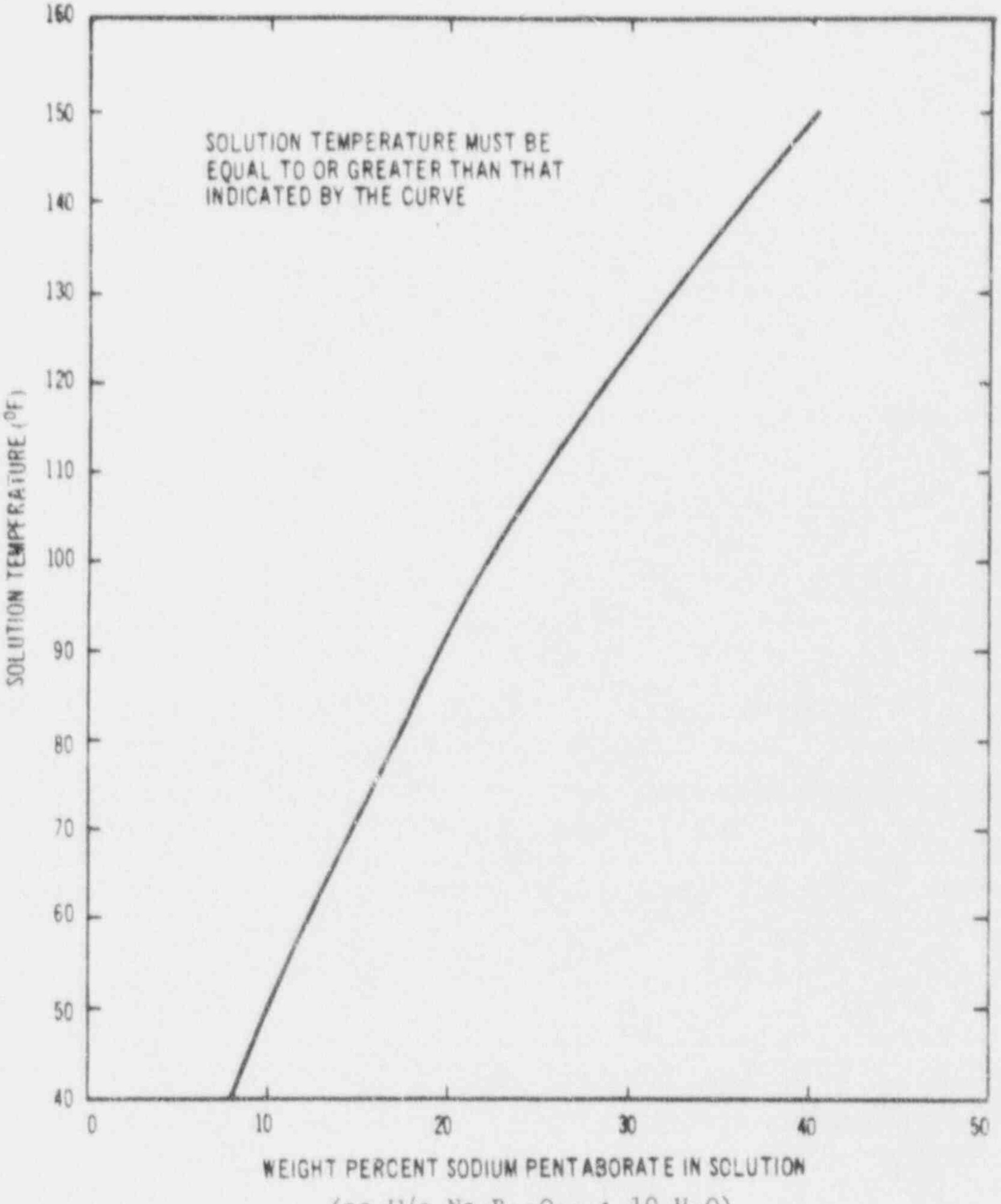


FIGURE 3.42 SODIUM PENTABORATE SOLUTION TEMPERATURE REQUIREMENTS

23 REV

Bases Continued 3.6 and 4.6:

D. Coolant Leakage

The former 15 gpm limit for leaks from unidentified sources was established assuming such leakage was coming from the primary system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. From the crack size a leakage rate can be determined. For a crack size which gives a leakage of 5 gpm, the probability of rapid propagation is less than $10^{-5}$. Thus, an unidentified leak of 5 gpm when assumed to be from the primary system had less than one chance in 100,000 of propagating, which provides adequate margin. A leakage of 5 gpm is detectable and measurable. The 24 hour period allowed for determination of leakage is also based on the low probability of the crack propagating.

The capacity of the drywell sump pumps is 100 gpm and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

An annual report will be prepared and submitted to the AEC summarizing the primary coolant to drywell leakage measurements. Other techniques for detecting leaks and the applicability of these techniques to the Monticello Plant will be the subject of continued study.

E. Safety and Relief Valves

Experience in safety valve operation shows that a testing of 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. A tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as $\pm 1\%$ of the set pressure. An analysis has been performed which shows that with all safety valves set 1% higher than the set pressure, the reactor coolant pressure safety limit of 1375 psig is not exceeded. Safety/relief valves are used to minimize activation of the safety valves. The operator will set the pressure settings at or below the settings listed. However, the actual set points can vary as listed in the basis of Specification 2.4.

The required safety valve steam flow capacity is determined by analyzing the pressure rise accompanying the main steam flow stoppage resulting from a MSIV closure with the reactor at 1670 MWt. The analysis assumes no MSIV closure scram, but a reactor scram from indirect means (high flux). The relief and safety valve capacity is assumed to total 83.9% (47% relief and 36.9% safety) of the full power steam generation rate. This capacity corresponds to assuming that four safety/relief valves (47%) and four safety valves (36.9%) operated.

| 3.0 LIMITING CONDITIONS FOR OPERATION | 4.0 SURVEILLANCE REQUIREMENTS |
|---|---|
| C. Secondary Containment | C. Secondary Containment |
| 1. Secondary containment integrity, shall be maintained during all modes of plant operation except when all of the following conditions are met. | 1. Secondary containment surveillance shall be performed as indicated below: |
| a. The reactor is subcritical and Specification 3.3.A is met. | a. Secondary containment capability to maintain at least a 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a filter train flow rate of ≤ 4,000 scfm, shall be demonstrated at each refueling outage prior to refueling. This surveillance testing should be reported in the semiannual operating reports. |
| b. The reactor water temperature is below 212° and the reactor coolant system is vented. | |
| c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A. | |

Bases Continued:

The acceptable values for local leak rate tests have been specified in terms of standard cubic feet per hour (scf/hr) for purposes of clarity. Following is the list of equivalent values given in terms of an allowable percentage of the allowable operational leak rate ($L_{to}$).

17.2 scf/hr = 5% $L_{to}$
@ 41 psig

34.4 scf/hr = 10% $L_{to}$
@ 41 psig

103.2 scf/hr = 30% $L_{to}$
@ 41 psig

where $L_{to}$ = .75 $L_t$ (the maximum allowable leak rate)
and $L_t$ = 1.2 weight percent of the contained air at the test pressure of 41 psig.

Results of loss of coolant accident analyses indicate that fission products would not be released directly to the environs because of leakage through the main line isolation valves due to holdup in the steam system complex. Although this effect shows that an adequate margin exists with regard to release of fission products, the results of leak tests on the main steam line isolation valves will be closely followed in order to determine the adequacy of these valves to perform their intended function. A summary report of the results of main steam line isolation valve leakage tests and closure time measurements will be prepared and submitted to the AEC following completion of periodic main steam line isolation valve leakage tests.

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time will be kept to a practical minimum.

(d) Highest, lowest, and the annual average concentrations or levels of radiation for the sampling point with the highest average and description of the location of that point with respect to the site.

(2) If levels of radioactive materials in environmental media as determined by an environmental monitoring program indicate the likelihood of public intakes in excess of 1% of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, Part 20, estimates of the likely resultant exposure to individuals and to population groups, and assumptions upon which estimates are based shall be provided.

(3) If statistically significant variation of offsite environmental concentrations with time are observed, correlation of these results with effluent release shall be provided.

i. Occupational Personnel Radiation Exposure

Tabulate the number of personnel exposures for plant personnel (permanent and temporary) in the following exposure increments for the reporting period:

less than 100 mrem, 100 - 500 mrem, 500 - 1250 mrem, 1250 - 2500 mrem, above 2500 mrem.

Tabulate the number of personnel receiving more than 500 mrem exposure in the reporting period according to duty function, i.e, routine plant surveillance and inspection (regular duty), routine plant maintenance, special plant maintenance (describe maintenance), routine refueling operations, special refueling operation (describe operation) and other job related exposures. Annually tabulate the number of personnel receiving more than 2500 mrem and report major cause(s).

B. Non-Routine Reports

1. Abnormal Occurrence Reports

Notification shall be made within 24 hours by telephone and telegraph to the Director of the Regional Regulatory Operations Office (cc to the Director of Licensing), followed by a written report within 10 days to the Director of Licensing (cc to the Director of the Regional Regulatory Operations Office) in the event of the abnormal occurrences as defined in Section 1.0. The written report on these abnormal occurrences, and to the extent possible, the preliminary telephone and telegraph notification, shall: (a) describe, analyze and evaluate safety implications, (b) outline the measures taken to assure that the cause of the condition is determined, (c) indicate the corrective action (including any changes made to the procedures and to the quality assurance program) taken to prevent repetition of the occurrence and of similar occurrences involving similar components or systems, and (d) evaluate the safety implications of the incident in light of the cumulative experience obtained from the record of previous failures and malfunctions of similar systems and components.

AEC DI^^TRIBUTION FOR PART 50 DOCKET MA^^ERIAL
(TEMPORARY FORM)

CONTROL NO: 7931

FILE:

| FROM: | DATE OF DOC | DATE REC'D | LTR | MEMO | RPT | OTHER |
|---|---|---|---|---|---|---|
| Northern States Power Company<br>Minneapolis, Minnesota 55401<br>L. O. Mayer | 10-26-73 | 10-31-73 | X | | | |

| TO: | ORIG | CC | OTHER | |
|---|---|---|---|---|
| J. F. O'Leary | 3 signed | | | SENT AEC PDR X<br>SENT LOCAL PDR X |

| CLASS | UNCLASS | PROP INFO | INPUT | NO CYS REC'D | DOCKET NO: |
|---|---|---|---|---|---|
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