ATTACHMENT A

PROPOSED CHANGES TO APPENDIX A,

TECHNICAL SPECIFICATIONS OF FACILITY

OPERATING LICENSES NPF-37, NPF-66, NPF-72 and NPF-75

Byron Station	Braidwood Station
Revised Pages: 3/4 4-27	Revised Pages: 3/4 4-27
3/4 4-28	3/4 4-28
3/4 4-40	3/4 4-40
3/4 5-1	3/4 5-1
3/4 6-23	----
3/4 7-14	----
B3/4 4-5	B3/4 4-5
B3/4 4-6	B3/4 4-6
5-6	5-6
6-7	----
6-8	6-8
6-13	6-13
6-18	6-18

REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross radioactivity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

~~a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12-month period. The provisions of Specification 3.0.4 are not applicable;~~

~~b. With the total cumulative operating time at a reactor coolant specific activity greater than 1 microCurie per gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6-month period, prepare and submit a Special Report to the Commission within 30 days, pursuant to Specification 6.9.2, indicating the number of hours above this limit. The provisions of Specification 3.0.4 are not applicable;~~

a. ~~c.~~ With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and

b. ~~d.~~ With the specific activity of the reactor coolant greater than $100/\bar{E}$ microCuries per gram of gross radioactivity, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

* With T_{avg} greater than or equal to 500°F.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ microCuries per gram of gross radioactivity, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, and Chief, Accident Evaluation Branch, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555. This report shall contain the results of the specific activity analyses together with the following information:

a. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded;

b. Results of: (1) the last isotopic analysis for radioiodine performed prior to exceeding the limit, (2) analysis while limit was exceeded, and (3) one analysis after the radioiodine was reduced to less than the limit, including for each isotopic analysis, the date and time of sampling and the radioiodine concentration;

c. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

d. History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

e. The time duration when the specific activity of the primary coolant exceeded 1 microCurie per gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

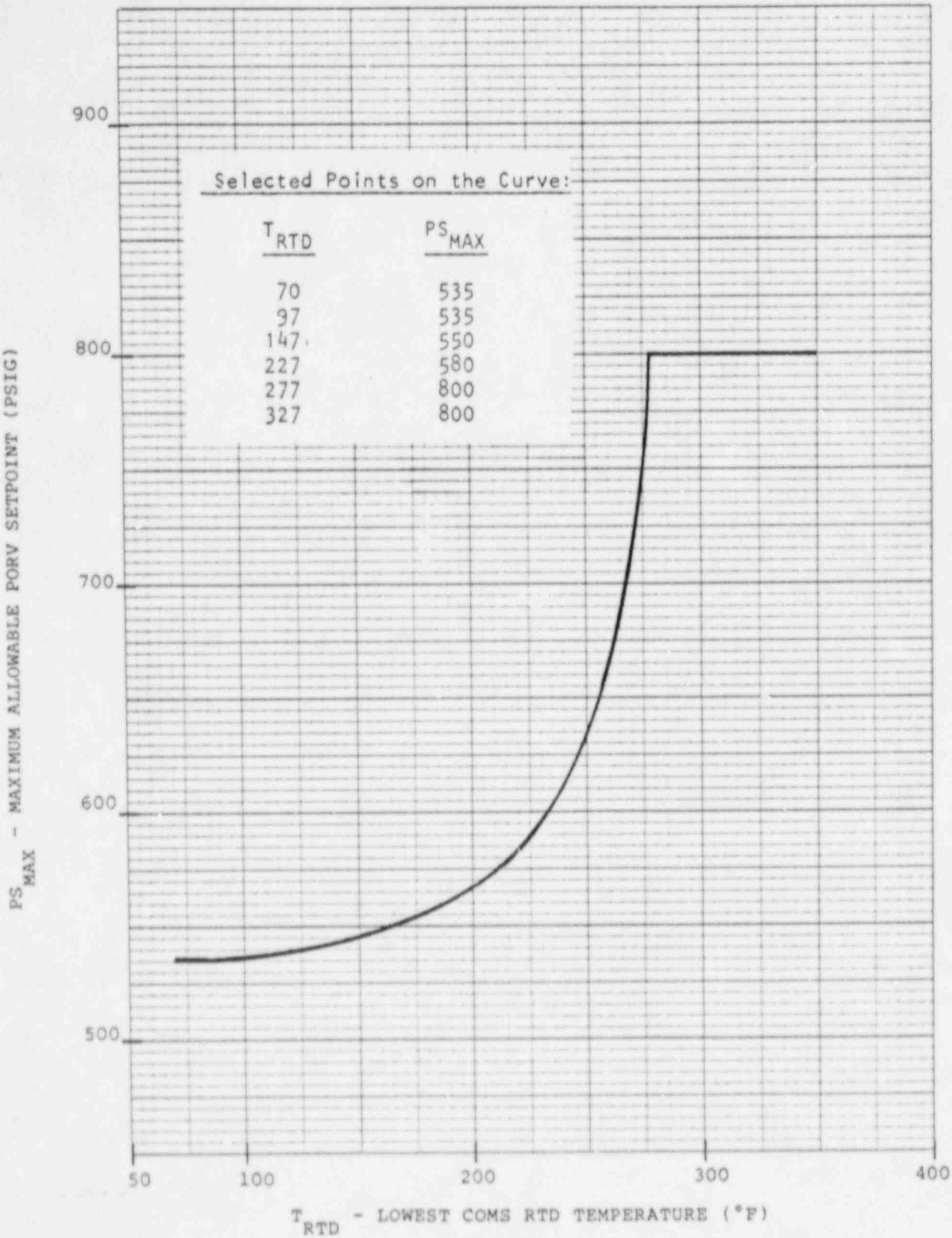

FIGURE 3.4-4

NOMINAL PORV PRESSURE RELIEF SETPOINT VERSUS
RCS TEMPERATURE FOR THE COLD OVERPRESSURE PROTECTION SYSTEM
APPLICABLE UP TO 10 EFPY

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water level of between 31% and 63%,

c. A boron concentration of between 1900 and 2100 ppm, and

d. A nitrogen cover-pressure of between 602 and 647 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, exc pt as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying ~~by the absence of alarms~~ the contained borated water level and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

CONTAINMENT ISOLATION VALVES

PENETRATION	VALVE NO.	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
9. Manual (Continued)			
99	FW015D*,#	Feedwater	N.A.
100	FW015A*,#	Feedwater	N.A.
101	FW015B*,#	Feedwater	N.A.
102	FW015C*,#	Feedwater	N.A.
10. Check			
28	CV8113	RCP Seal Water Return	N.A.
37	CV8348*	RCS Loop Fill	N.A.
6	WØ007A	Chilled Water	N.A.
10	WØ007B	Chilled Water	N.A.
21	CC9534	RCP Mtr Brng Return	N.A.
24	CC9518	RCP Thermal Barrier Return	N.A.
25	CC9486	RCP Cooling Wtr Supply	N.A.
1	CS008A	Containment Spray	N.A.
16	CS008B	Containment Spray	N.A.
39	IA091	Instrument Air	N.A.
30	WM191	Make-Up Demin	N.A.
52	PR032	Process Radiation	N.A.
AL	PR002G	Process Radiation	N.A.
AL	PR002H	Process Radiation	N.A.
12	PS231A	Hydrogen Monitor	N.A.
31	PS231B	Hydrogen Monitor	N.A.
27	RY8047	PRT Nitrogen	N.A.
44	RY8046	PRT Make-Up	N.A.
26	SI8815*	Safety Injection	N.A.
50	SI8818A*	Safety Injection	N.A.
50	SI8818D*	Safety Injection	N.A.
51	SI8818B*	Safety Injection	N.A.
51	SI8818C*	Safety Injection	N.A.
59	SI8905A*	Safety Injection	N.A.
59	SI8105D* ~~SI8905D*~~	Safety Injection	N.A.
60	SI8819A*	Safety Injection	N.A.
60	SI8819B*	Safety Injection	N.A.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. With one essential service water makeup pump inoperable, restore the essential service water makeup pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With the essential service water pump discharge water temperature not meeting the above requirement, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With the minimum Rock River water level not meeting the above requirement, notify the NRC within 1 hour in accordance with the procedure of 10 CFR 50.72 of actions or contingencies to ensure an adequate supply of cooling water to the Byron Station for a minimum of 30 days, verify the Rock River flow within 1 hour, and:

(1) If Rock River flow is less than 700 cubic feet per second (cfs) be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours, or

(2) If Rock River flow is equal to or greater than 700 cfs continue verification procedure every 12 hours or until Rock River water level exceeds 670.6 feet MSL or

(3) If Rock River level is equal to or less than 664.7 feet MSL be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours

f. With one deep well inoperable and:

(1) The Rock River water level predicted, through National Weather Service flood forecasts, to exceed 702 feet MSL, or

(2) The Rock River water level at or below 670.6 feet MSL, or

(3) A tornado watch issued by the NWS that includes the area for the Byron Station.

Notify the NRC within 1 hour in accordance with the procedure of 10 CFR 50.72 of actions or contingencies to ensure an adequate supply of cooling water to the Byron Station for a minimum of 30 days and restore both wells to OPERABLE status before the Rock River water level exceeds 702 feet MSL or the minimum Rock River level or flow falls below 664.7 feet MSL or 700 cfs, respectively, or within 72 hours, whichever occurs first, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 The UHS shall be determined OPERABLE at least once per:

a. 24 hours by verifying the water level in each UHS cooling tower basin to be greater than or equal to ~~873.5~~ feet MSL. (50%),

873.75

OPERATIONAL LEAKAGE (Continued)

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Byron Station, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

~~The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 1 microCurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1 microCurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10% of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the SITE BOUNDARY by a factor of up to 20 following a postulated steam generator tube~~

SPECIFIC ACTIVITY (Continued)

~~rupture. The reporting of cumulative operating time over 500 hours in any 6-month consecutive period with greater than 1 microCurie/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800-hour limit.~~

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would be about 20%. The exclusion of radionuclides with half-lives less than 10 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinct window for determination of the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample with typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at $\leq$ 100°F/h and 200 cooldown cycles at $\leq$ 100°F/h.	Heatup cycle - T_{avg} from $\leq$ 200°F to $\geq$ 550°F. Cooldown cycle - T_{avg} from $\geq$ 550°F to $\leq$ 200°F.
	200 pressurizer cooldown cycles at $\leq$ 200°F/h.	Pressurizer cooldown cycle temperatures from $\geq$ 650°F to $\leq$ 100°F.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	$\geq$ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray actuation cycles.	Spray water temperature differential > 320°F.
	200 ~~50~~ leak tests.	Pressurized to $\geq$ 2485 psig.
	10 ~~5~~ hydrostatic pressure tests.	Pressurized to $\geq$ 3107 ~~3100~~ psig.
Secondary Coolant System	1 large steam line break.	Break in a > 6-inch steam line.
	10 ~~5~~ hydrostatic pressure tests.	Pressurized to $\geq$ 1481 ~~1350~~ psig.

6.5 REVIEW INVESTIGATION AND AUDIT (Continued)

OFFSITE

6.5.1 The Superintendent of the Offsite Review and Investigative Function shall be appointed by the Manager of Nuclear Safety responsible for nuclear activities. The audit function shall be the responsibility of the Manager of Quality Assurance and shall be independent of operations.

a. Offsite Review and Investigative Function

The Superintendent of the Offsite Review and Investigative Function shall: (1) provide directions for the review and investigative function and appoint a senior participant to provide appropriate direction, (2) select each participant for this function, (3) select a complement of more than one participant who collectively possess background and qualifications in the subject matter under review to provide comprehensive interdisciplinary review coverage under this function, (4) independently review and approve the findings and recommendations developed by personnel performing the review and investigative function, (5) approve and report in a timely manner all findings of non-compliance with NRC requirements to the Station Manager, Assistant Vice President and General Manager - Nuclear Stations, Manager of Quality Assurance, and the Vice President - Nuclear Operations. During periods when the Supervisor of Offsite Review and Investigative Function is unavailable, he shall designate this responsibility to an established alternate, who satisfies the formal training and experience for the Superintendent of the Offsite Review and Investigative Function. The responsibilities of the personnel performing this function are stated below. The Offsite Review and Investigative Function shall review: Superintendent

1) The safety evaluations for: (1) changes to procedures, equipment, or systems as described in the safety analysis report, and (2) tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question. Proposed changes to the Quality Assurance Program description shall be reviewed and approved by the Manager of Quality Assurance;

2) Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

3) Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

4) Proposed changes in Technical Specifications or this Operating License;

OFFSITE (Continued)

5) Noncompliance with Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures, or instructions having nuclear safety significance;

6) Significant operating abnormalities or deviation from normal and expected performance of plant equipment that affect nuclear safety as referred to it by the Onsite Review and Investigative Function;

7) All REPORTABLE EVENTS;

8) All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components;

9) Review and report findings and recommendations regarding all changes to the Generating Stations Emergency Plan prior to implementation of such change; and

10) Review and report findings and recommendations regarding all items referred by the Technical Staff Supervisor, Station Manager, Assistant Vice President and General Manager - Nuclear Stations, and Manager of Quality Assurance. 2

of the above, or designated Corporate Staff or Supervision approved by the Manager of Quality Assurance

b. Audit Function

The audit function shall be the responsibility of the Manager of Quality Assurance independent of the Production Department. Such responsibility is delegated to the Director of Quality Assurance (Operations) and the Director of Quality Assurance (Maintenance). 2

Either shall approve the audit agenda and checklists, the findings and the report of each audit. Audits shall be performed in accordance with the Company Quality Assurance Program and Procedures. Audits shall be performed to assure that safety-related functions are covered within the period designated below:

1) The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

2) The adherence to procedure, training, and qualification of the station staff at least once per 12 months;

3) The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or methods of operation that affect nuclear safety at least once per 6 months;

4) The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

ONSITE (Continued)

3) Review of all proposed changes to the Technical Specifications;

4) Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety;

5) Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Assistant Vice President and General Manager - Nuclear Stations and to the Superintendent of the Offsite Review and Investigative Function; 8

6) Review of all REPORTABLE EVENTS;

7) Performance of special reviews and investigations and reports thereon as requested by the Superintendent of the Offsite Review and Investigative Function; 8

8) Review of the Station Security Plan and implementing ~~procedures~~ and submittal of recommended changes to the ~~Assistant Vice President and General Manager - Nuclear Stations;~~ Station Security Plan to the Director of Corporate Security;

9) Review of the Emergency Plan and station implementing procedures and submittal of recommended changes to the Assistant Vice President and General Manager - Nuclear Stations;

10) Review of Unit operations to detect potential hazards to nuclear safety;

11) Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Assistant Vice President and General Manager - Nuclear Stations and the Superintendent of the Offsite Review and Investigative Function; and 8

12) Review of changes to the PROCESS CONTROL PROGRAM, the OFFSITE DOSE CALCULATION MANUAL, and the Radwaste Treatment Systems.

13) Review of the Fire Protection Program and implementing instructions and submittal of recommended changes to the Offsite Review and Investigative Function.

c. Authority

The Technical Staff Supervisor is responsible to the Station Manager and shall make recommendations in a timely manner in all areas of review, investigation, and quality control phases of plant maintenance, operation, and administrative procedures relating to facility operations and shall have the authority to request the action necessary to ensure compliance with rules, regulations, and procedures when in his opinion such action is necessary. The Station Manager shall follow such recommendations or select a course

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

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

STARTUP REPORT

6.9.1.1 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.

6.9.1.2 The Startup Report shall address each of the tests identified in the Final Safety Analysis Report FSAR and shall 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.

6.9.1.3 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 operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS

6.9.1.4 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include: a. Tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrems/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 assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling 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 should be assigned to specific major work functions.

b. See Insert "A"

*This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross radioactivity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

~~a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12-month period. The provisions of Specification 3.0.4 are not applicable;~~

~~b. With the total cumulative operating time at a reactor coolant specific activity greater than 1 microCurie per gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6-month period, prepare and submit a Special Report to the Commission within 30 days, pursuant to Specification 6.9.2, indicating the number of hours above this limit. The provisions of Specification 3.0.4 are not applicable;~~

a. ~~c.~~ With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and

b. ~~d.~~ With the specific activity of the reactor coolant greater than $100/\bar{E}$ microCuries per gram ~~of gross radioactivity~~, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

* With T_{avg} greater than or equal to 500°F.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1 microCurie ~~per gram DOSE EQUIVALENT~~ I-131 or greater than 100/Ē microCuries per gram ~~of gross radioactivity~~, perform the sampling and analysis requirements of ~~Item~~ 4.a) of Table ~~4.4-4~~ until the ~~specific activity of the reactor coolant~~ is restored to within its limits. ~~Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days with a copy to the Director, Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555. This report shall contain the results of the specific activity analyses together with the following information:~~

a. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded;

b. Results of: (1) the last isotopic analysis for radioiodine performed prior to exceeding the limit, (2) analysis while limit was exceeded, and (3) one analysis after the radioiodine was reduced to less than the limit, including for each isotopic analysis, the date and time of sampling and the radioiodine concentration;

c. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

d. History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

e. The time duration when the specific activity of the primary coolant exceeded 1 microCurie per gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

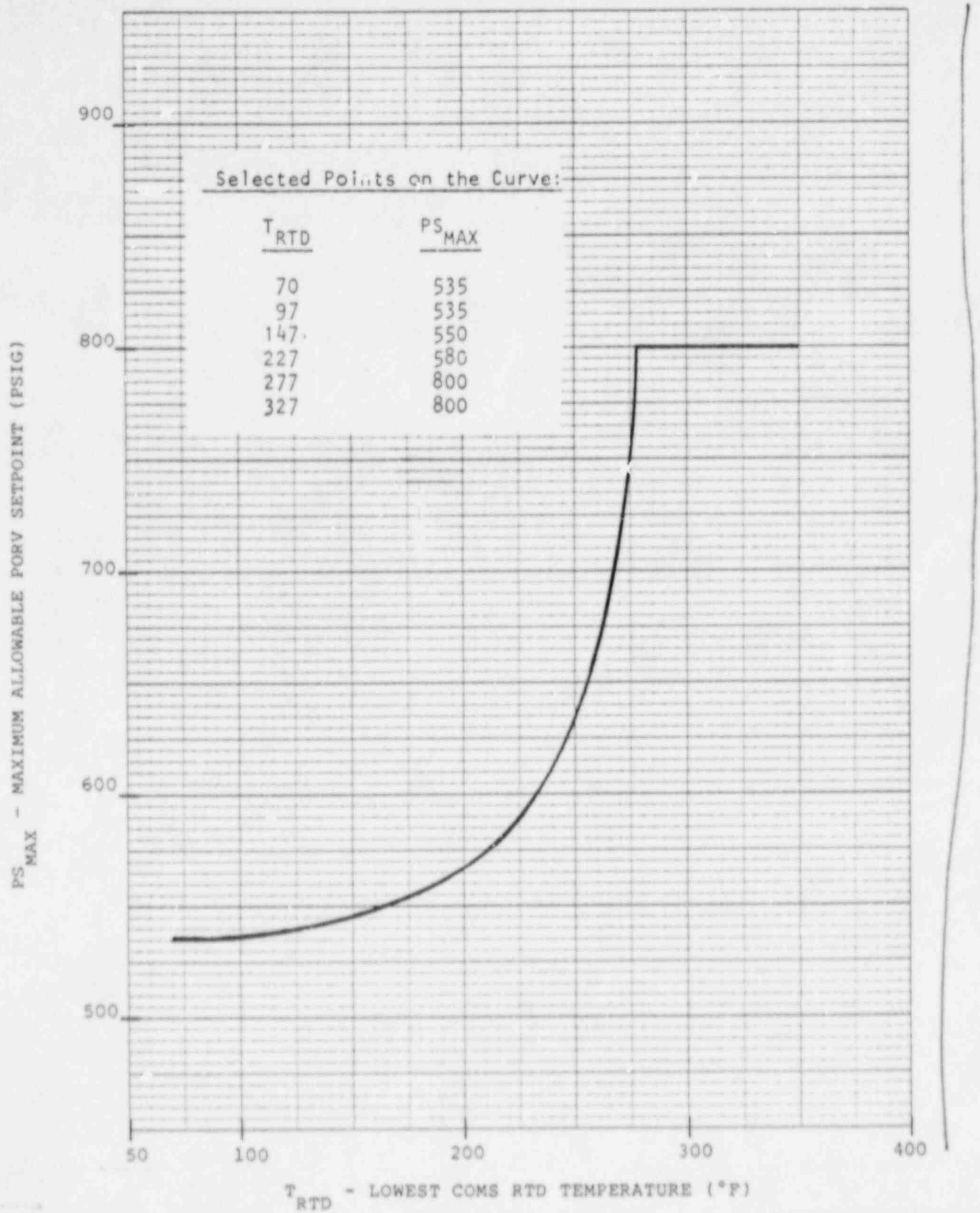

FIGURE 3.4-4

NOMINAL PORV PRESSURE RELIEF SETPOINT VERSUS
RCS TEMPERATURE FOR THE COLD OVERPRESSURE PROTECTION SYSTEM
APPLICABLE UP TO 10 EFPY

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water level of between 31% and 63%,

c. A boron concentration of between 1900 and 2100 ppm, and

d. A nitrogen cover-pressure of between 602 and 647 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, ~~by the absence of alarms,~~ the contained borated water level and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

OPERATIONAL LEAKAGE (Continued)

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Braidwood Station, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

~~The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 1 microCurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1 microCurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 100 hours per year (approximately 10% of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the SITE BOUNDARY by a factor of up to 20 following a postulated steam generator tube~~

SPECIFIC ACTIVITY (Continued)

~~rupture. The reporting of cumulative operating time over 500 hours in any 6-month consecutive period with greater than 1 microCurie/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800-hour limit.~~

The sample analysis for determining the gross specific activity and Ē can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would be about 20%. The exclusion of radionuclides with half-lives less than 10 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinct window for determination of the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample with typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100°F/h and 200 cooldown cycles at ≤ 100°F/h.	Heatup cycle - T_{avg} from ≤ 200°F to ≥ 550°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.
	200 pressurizer cooldown cycles at ≤ 200°F/h.	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 100°F.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	≥ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray actuation cycles.	Spray water temperature differential > 320°F.
	200 ~~50~~ leak tests.	Pressurized to ≥ 2485 psig.
	10 ~~5~~ hydrostatic pressure tests.	Pressurized to ≥ 3107 ~~3100~~ psig.
Secondary Coolant System	1 large steam line break.	Break in a > 6-inch steam line.
	10 ~~5~~ hydrostatic pressure tests.	Pressurized to ≥ 1481 ~~1350~~ psig.

OFFSITE (Continued)

5) Noncompliance with Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures, or instructions having nuclear safety significance;

6) Significant operating abnormalities or deviation from normal and expected performance of plant equipment that affect nuclear safety as referred to it by the Onsite Review and Investigative Function;

7) All REPORTABLE EVENTS;

8) All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components;

9) Review and report findings and recommendations regarding all changes to the Generating Stations Emergency Plan prior to implementation of such change; and

10) Review and report findings and recommendations regarding all items referred by the Technical Staff Supervisor, Station Manager, Assistant Vice President and General Manager of Nuclear Stations~~, and Manager of Quality Assurance~~.

b. Audit Function

of the above, or designated Corporate Staff or Supervision approved by the Manager of Quality Assurance

The audit function ~~shall be the responsibility~~ of the Manager of Quality Assurance independent of the Production Department. Such responsibility is delegated to the Director of Quality Assurance (Operations) and the Director of Quality Assurance (Maintenance).

Either shall approve the audit agenda and checklists, the findings and the report of each audit. Audits shall be performed in accordance with the Company Quality Assurance Program and Procedures. Audits shall be performed to assure that safety-related functions are covered within the period designated below:

1) The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

2) The adherence to procedure, training, and qualification of the station staff at least once per 12 months;

3) The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or methods of operation that affect nuclear safety at least once per 6 months;

4) The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

ONSITE (Continued)

3) Review of all proposed changes to the Technical Specifications;

4) Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety;

5) Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Assistant Vice President and General Manager of Nuclear Stations and to the Superintendent of the Offsite Review and Investigative Function;

6) Review of all REPORTABLE EVENTS;

7) Performance of special reviews and investigations and reports thereon as requested by the Superintendent of the Offsite Review and Investigative Function;

8) Review of the Station Security Plan and implementing procedures and submittal of recommended Security Plan changes to the Director of Corporate Security;

9) Review of the Emergency Plan and station ~~implementing procedures~~ and submittal of recommended changes to the ~~Assistant Vice~~ Station ~~President and General Manager of Nuclear Stations~~; Security Plan to the Director of Corporate Security

10) Review of Unit ~~operations~~ to ~~detect~~ potential ~~hazards~~ to ~~nuclear~~ safety;

11) Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Assistant Vice President and General Manager of Nuclear Stations and the Superintendent of the Offsite Review and Investigative Function; and

12) Review of changes to the PROCESS CONTROL PROGRAM, the OFFSITE DOSE CALCULATION MANUAL, and the Radwaste Treatment Systems.

13) Review of the Fire Protection Program and implementing instructions and submittal of recommended changes to the Offsite Review and Investigative Function.

c. Authority

The Technical Staff Supervisor is responsible to the Station Manager and shall make recommendations in a timely manner in all areas of review, investigation, and quality control phases of plant maintenance, operation, and administrative procedures relating to facility operations and shall have the authority to request the action necessary to ensure compliance with rules, regulations, and procedures when in his opinion such action is necessary. The Station Manager shall follow such recommendations or select a course of action that

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

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

STARTUP REPORT

6.9.1.1 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.

6.9.1.2 The Startup Report shall address each of the tests identified in the Final Safety Analysis Report FSAR and shall 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.

6.9.1.3 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 operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS

6.9.1.4 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include: a. Tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrems/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 assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling 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 should ~~be assigned to specific major~~ work functions.

b. See Insert "A"

*This tabulat~~ion supplements~~ the requirements of §20.407 of 10 CFR Part 20.

INSERT "A"

The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

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ATTACHMENT B

DESCRIPTION AND SUMMARY OF PROPOSED CHANGES

The proposed changes involve the following Technical Specifications for the Byron and Braidwood Stations:

1) Specific Activity, Specifications 3.4.8, Bases 3/4.4.8, and 6.9.1.5 (pages 3/4 4-27 and 28, B 3/4 4-5 and 6, and 6-18)

   This change is requested based on Generic Letter 85-19, dated September 27, 1985. The subject of the Generic Letter involved the reporting requirement on primary coolant iodine spikes. The Technical Specification changes requested are consistent with those proposed in the Generic Letter.

2) Cold Overpressure Protection System Setpoints, Figure 3.4-4, (page 3/4 4-40)

   This change is requested based on a letter from Westinghouse indicating two changes should be made to the current Cold Overpressure Protection system curve in the Technical Specifications. The first change is to accomodate a larger wide range temperature uncertainty than had been previously utilized. Secondly, the revised Figure eliminates the need for a detailed evaluation of resultant stresses in the PORV inlet and discharge piping and steam generator tube sheet following a single overpressure event. The proposed change to Figure 3.4-4 reflects these revised setpoints.

3) Accumulators, Specification 4.5.1.1.a.1 (page 3/4 5-1)

   This change is requested to delete the words "by the absence of alarms", since the current wording could be interpreted that the Unit must be shutdown if an annunciator failed, even though other indications available to the operators would verify that the Technical Specification limits are being met. Deleting these words allows the operators to verify the accumulator borated water level and nitrogen cover pressure by using the safety-related level and pressure instruments that are periodically calibrated.

4) Containment Isolation Valves, Table 3.6-1 (Page 3/4 6-23)

   This change is requested to correct a typographical error for one Safety Injection Valve number, from "SI 8805D" to "SI 8905D". This is a Byron Station only change since this has been corrected in the Braidwood Technical Specifications, previously.

5) Plant Systems, Specification 4.7.5 (Page 3/4 7-14)

This change is requested to correct a typographical error of the value of the UHS cooling tower basin water level from "873.5 feet" to "873.75 feet". This change is for Byron Station only because of the site specific feature of this Technical Specification.

6) Component Cyclic or Transient Limits, Table 5.7-1 (page 5-6)

This change is requested to make the number of design transients, assumed for the primary side leak test and the primary and secondary hydrostatic pressure tests, consistent with the number of design transients listed in Byron/Braidwood FSAR Table 3.9-1. In addition, a change is requested for the primary and secondary hydrostatic pressure test design limits, since this value should be 1.25 times the design pressure as required in ASME Section XI.

7) Administrative Controls, Specification 6.5 (Pages 6-7, 6-8 and 6-13)

This change is requested to correct and update the titles of various Commonwealth Edison management personnel based upon the current company organization and provide a clearer description of whom has authority to approve Quality Assurance audit agenda and checklists. The change, requested for page 6-7, has been corrected in the Braidwood Technical Specification, previously.

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

Commonwealth Edison has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
3) Involve a significant reduction in a margin of safety.

All the changes requested have been evaluated as presented below:

1) The changes to the Specific Activity Technical Specifications 3.4-8, Bases 3/4.4.8, and 6.9.1.5 are consistent with Generic Letter 85-19. The reporting requirements for iodine spiking are being reduced from a Special Report to an item to be included in an Annual Report. Also, the requirement to shutdown the plant if coolant iodine activity limits are exceeded for 800 hours in a 12 month period has been deleted because of improved industry wide fuel quality whereby normal coolant iodine activity is well below this limit. In addition, 10 CFR 50.72(b)(1)(ii) requires the NRC to be immediately notified of fuel cladding failures that exceed expected values or that are caused by unexpected factors. Therefore, this Technical Specification limit is no longer considered necessary on the basis that proper fuel management and existing reporting requirements should preclude ever approaching the limit. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

   Since the changes are consistent with the appropriate Byron/Braidwood FSAR section and analysis and no physical modifications are being made in the plant, the possibility for an accident or malfunction of a different type, than any previously evaluated is not created.

   Since appropriate measures will remain in place to address primary coolant iodine spiking, the margin of safety will not be reduced.

2) The Cold Overpressure Protection (COP) system setpoints (page 3/4 4-40) in the current Technical Specifications and those requested both meet the Appendix G criteria. The changes request more conservative COP system setpoints to address a larger uncertainty assumed in the wide range temperature instrumentation and to prevent the need for additional stress evaluations following a single overpressure event. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Since the changes are consistent with the appropriate Byron/Braidwood FSAR section and analysis and no physical modifications are being made in the plant, the possibility for an accident or malfunction of a different type, than any previously evaluated is not created.

Since the cold overpressure protection response setpoints are more conservative than the current Technical Specifications the margin of safety does not involve a significant reduction.

3) Accumulator Technical Specification 4.5.1.1.9.1 will still require verification of accumulator parameters assumed in the Byron/Braidwood FSAR analysis but the revised wording will allow the operators flexibility in how these parameters will be verified. The changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Since the changes are consistent with the appropriate Byron/Braidwood FSAR section and analysis and no physical modifications are being made in the plant, the possibility for an accident or malfunction of a different type, than any previously evaluated is not created.

Since the appropriate administrative controls will remain in place to verify accumulator parameters assumed in the Byron/Braidwood FSAR, the margin of safety is not reduced.

4) The change to Table 3.6-1 is being made to correct a typographical error for one Safety Injection Valve number. The change does not involve an increase in the probability or consequences of an accident previously evaluated.

Since the changes are consistent with the appropriate Byron/Braidwood FSAR section and analysis and no physical modifications are being made in the plant, the possibility for an accident or malfunction of a different type than any previously is not created.

Since the Table 3.6-1 changes are being made to be consistent with the Byron/Braidwood FSAR, the margin of safety is not reduced.

5) The change to Technical Specification 4.7.5 is being made to correct a typographical error for the UHS cooling tower basin water level. The change does not involve a significant increase in the probability or consequence of an accident previously evaluated.

Since the changes are consistent with the appropriate Byron/Braidwood FSAR section and analysis and no physical modifications are being made in the plant, the possibility for an accident or malfunction of a different type than any previously evaluated is not created.

Since the changes are being made to be consistent with the Byron/Braidwood FSAR, the margin of safety is not reduced.

6) The changes to Table 5.7-1 are being made to make the values be consistent with the Byron/Braidwood FSAR. The changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Since the changes are consistent with the appropriate Byron/Braidwood FSAR section and analysis and no physical modifications are being made in the plant, the possibility for an accident or malfunction of a different type than any previously evaluated is not created.

Since the Table 5.7-1 changes are being made to be consistent with the Byron/Braidwood FSAR, the margin of safety is not reduced.

Therefore, based on the above evaluations, Commonwealth Edison believes that these changes do not involve significant hazards considerations.

7) The changes to technical specification 6.5 are administrative in nature and are being made to clarify some Edison management titles and to further describe the functional authority of some Edison Quality Assurance personnel. The change does not involve a significant increase in the probability or consequence of an accident previously evaluated.

Since the changes are consistent with the appropriate Byron/Braidwood FSAR section and analysis and no physical modifications are being made in the plant, the possiblity for an accident or malfunction of a different type, than any previously evaluated is not created.

Since the appropriate administrative controls will remain in place and are not being changed, the margin of safety is not reduced.

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