Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515-5701
www.exeloncorp.com

ComEd
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10 CFR 50.90

RS-00-0167

December 27, 2000

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Request for License Amendment for Power Uprate Operation

References:

1) Licensing Topical Report, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32424P-A, Class III, February 1999.

2) Licensing Topical Report, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32523P-A, Class III, February 2000, and Supplement 1, Volumes I and II.

3) Letter from U.S. NRC to G.L. Sozzi (General Electric), "Staff Position Concerning General Electric Boiling-Water Reactor Extended Power Uprate Program," dated February 8, 1996

4) Letter from U.S. NRC to J.F. Quirk (General Electric), "Staff Safety Evaluation of General Electric Boiling Water Reactor (BWR) Extended Power Uprate Generic Analyses," dated September 14, 1998

5) Letter from U.S. NRC to R.O. Anderson (Northern States Power), "Issuance of Amendment Re: Power Uprate Program," dated September 16, 1998

6) Letter from U.S. NRC to H.L. Sumner, Jr. (Southern Nuclear Operating Company), "Issuance of Amendments – Edwin I. Hatch Nuclear Plant, Units 1 and 2," dated October 22, 1998

APO1

7) Letter from R. M. Krich (ComEd) to U.S. NRC, "Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated March 3, 2000

8) Letter from R.M. Krich (Commonwealth Edison Company) to U.S. NRC, "Request for Technical Specifications Change, Transition to General Electric Fuel," dated September 29, 2000

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Commonwealth Edison (ComEd) Company requests changes to Facility Operating Licenses DPR-19, DPR-25, DPR-29, and DPR-30 and Appendix A to the Operating Licenses, the Technical Specifications (TS), for Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. The proposed changes will allow the DNPS and QCNPS units to operate at an uprated power level of 2957 megawatts thermal (MWt). This represents an increase of approximately 17 percent rated core thermal power over the current 100 percent power level of 2527 MWt for DNPS and an increase of approximately 17.8 percent rated core thermal power over the current 100 percent power level of 2511 MWt for QCNPS.

The proposed changes follow the generic guidelines for uprating the power of Boiling Water Reactors (BWRs) described in References 1 and 2 and approved by the NRC in References 3 and 4. The proposed changes are similar to changes approved for Monticello Nuclear Generating Plant and Edwin I. Hatch Nuclear Plant in References 5 and 6.

In Reference 7, DNPS and QCNPS submitted TS amendment requests for conversion to the Improved Technical Specifications (ITS). In anticipation of approval of that request, this request for amendment is based on the format of the ITS.

In reference 8, DNPS and QCNPS submitted TS amendment requests to support a change in fuel vendors from Siemens Power Corporation to General Electric (GE) and a transition to GE 14 fuel. In anticipation of approval of that request, the evaluations supporting this amendment request were conducted for GE 14 fuel.

Outage-related modifications to support the implementation of these proposed changes will be made during the next planned refueling outages. Other modifications will be implemented prior to operating at uprated conditions. ComEd plans to fully implement the uprated power conditions for DNPS during the refueling outages scheduled to begin October 20, 2001, and September 28, 2002, for Units 2 and 3 respectively. ComEd plans to fully implement the uprated power conditions for QCNPS during the refueling outages scheduled to begin October 26, 2002, and February 2, 2002, for Units 1 and 2 respectively. Therefore, ComEd requests that these proposed changes be approved by October 15, 2001. ComEd also requests that the required implementation date for each unit be specified as prior to startup from the respective refueling outages.

This amendment request contains separate attachments for DNPS and QCNPS. Each attachment is subdivided as follows.

1. Attachment A contains a detailed description of the specific proposed changes necessary for operation at uprated conditions and the technical bases for these changes.
2. Attachment B provides the proposed markups to the TS.
3. Attachment C provides the information supporting a finding of no significant hazards consideration in accordance with 10 CFR 50.92(c), "Issuance of Amendment."
4. Attachment D provides supplements to the DNPS and QCNPS environmental reports, in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring an environmental assessment."
5. Attachment E contains the detailed plant-specific safety analysis required by the generic guidelines. This enclosure contains proprietary information and we request that it be withheld from public disclosure in accordance with 10 CFR 2.790(a)(4), "Public Inspections, Exemptions, Requests for Withholding."
6. Attachment F contains the affidavit supporting the request for withholding Attachment E from public disclosure, as required by 10 CFR 2.790(b)(1).
7. Attachment G describes the plant modifications required to support power uprate.

The proposed changes have been reviewed by the Plant Operations Review Committees and the Nuclear Safety Review Boards at DNPS and QCNPS in accordance with the Quality Assurance Program.

ComEd is notifying the State of Illinois of this license amendment request by transmitting a copy of this letter and its attachments to the designated State Official.

Should you have any questions related to this request, please contact Mr. Allan R. Haeger at (630) 663-6645

Respectfully,

R.M. Krich
Director, Licensing
Mid-West Regional Operating Group

Attachments

Affidavit

(Separate Attachments for DNPS and QCNPS)
Attachment A: Description and Summary Safety Analysis for Proposed Changes
Attachment B: Marked-Up TS Pages for Proposed Changes
Attachment C: Information Supporting a Finding of No Significant Hazards Consideration
Attachment D: Supplement to DNPS/QCNPS Environmental Report

Attachment E: GE Report NEDC-32962P, "Power Uprate Safety Analysis Report for Dresden Nuclear Power Station Units 2 and 3," December 2000 (Proprietary)
GE Report NEDC-32961P, "Power Uprate Safety Analysis Report for Quad Cities Nuclear Power Station Units 1 and 2," December 2000 (Proprietary)

Attachment F: GE Affidavit for withholding NEDC-32961P and NEDC-32962P from public disclosure

Attachment G: Plant Modifications Required to Support Power Uprate

cc:
Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Dresden Nuclear Power Station
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

STATE OF ILLINOIS)

COUNTY OF DUPAGE)

IN THE MATTER OF:)

COMMONWEALTH EDISON (COMED) COMPANY) Docket Numbers

DRESDEN NUCLEAR POWER STATION - Units 2 and 3) 50-237 and 50-249
QUAD CITIES NUCLEAR POWER STATION – Units 1 and 2) 50-254 and 50-265

SUBJECT: Request for License Amendment for Power Uprate Operation

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

R. M. Krich
Director, Licensing
Mid-West Regional Operating Group

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this 27th day of

December, 2000

"OFFICIAL SEAL"
Timothy A. Byam
Notary Public, State of Illinois
My Commission Expires 11/24/2001

Notary Public

GE AFFIDAVIT FOR NEDC-32962P

General Electric Company

AFFIDAVIT

I, George B. Stramback, being duly sworn, depose and state as follows:

(1) I am Project Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary report NEDC-32962P, *Safety Analysis Report for Dresden 2 & 3 Extended Power Uprate,* Class III (GE Proprietary Information), dated December 2000. This document, taken as a whole, constitutes a proprietary compilation of information, some of it also independently proprietary, prepared by the General Electric Company. The independently proprietary elements are identified by bars marked in the left margin adjacent to the specific material.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

Both the compilation as a whole and the marked independently proprietary elements incorporated in that compilation are considered proprietary for the reason described in items (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. That information (both the entire body of information in the form compiled in this document, and the marked individual proprietary elements) is of a sort customarily held in confidence by GE, and has, to the best of my knowledge, consistently been held in confidence by GE, has not been publicly disclosed, and is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified by bars in the margin is classified as proprietary because it contains detailed results and conclusions from these evaluations, utilizing analytical models and methods, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of transient and accident events in the GE Boiling Water Reactor ("BWR"). The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several million dollars.

The remainder of the information identified in paragraph (2), above, is classified as proprietary because it constitutes a confidential compilation of information, including detailed results of analytical models, methods, and processes, including computer codes, and conclusions from these applications, which represent, as a whole, an integrated process or approach which GE has developed, obtained NRC approval of, and applied to perform evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability of a given increase in licensed power output for a GE BWR. The development and approval of this overall approach was achieved at a significant additional cost to GE, in excess of a million dollars, over and above the very large cost of developing the underlying individual proprietary analyses.

To effect a change to the licensing basis of a plant requires a thorough evaluation of the impact of the change on all postulated accident and transient events, and all other regulatory requirements and commitments included in the plant's FSAR. The analytical process to perform and document these evaluations for a proposed power uprate was developed at a substantial investment in GE resources and expertise. The results from these evaluations identify those BWR systems and components, and those postulated events, which are impacted by the changes required to accommodate operation at increased power levels, and, just as importantly, those which are <u>not</u> so impacted, and the technical justification for not considering the latter in changing the licensing basis. The scope thus determined forms the basis for GE's offerings to support utilities in both performing analyses and providing licensing consulting services. Clearly, the scope and magnitude of effort of any attempt by a competitor to effect a similar licensing change can be narrowed considerably based upon these results. Having invested in the initial evaluations and developed the solution strategy and process described in the subject document GE derives an important competitive advantage in selling and performing these services. However, the mere knowledge of the impact on each system and component reveals the process, and provides a guide to the solution strategy.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive

physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods, including justifications for not including certain analyses in applications to change the licensing basis.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to avoid fruitless avenues, or to normalize or verify their own process, or to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions. In particular, the specific areas addressed by any document and submittal to support a change in the safety or licensing bases of the plant will clearly reveal those areas where detailed evaluations must be performed and specific analyses revised, and also, by omission, reveal those areas not so affected.

While some of the underlying analyses, and some of the gross structure of the process, may at various times have been publicly revealed, enough of both the analyses and the detailed structural framework of the process have been held in confidence that this information, in this compiled form, continues to have great competitive value to GE. This value would be lost if the information as a whole, in the context and level of detail provided in the subject GE document, were to be disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources, including that required to determine the areas that are not affected by a power uprate and are therefore blind alleys, would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing its analytical process.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

George B. Stramback, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 20th day of December 2000.

George B. Stramback

George B. Stramback
General Electric Company

Subscribed and sworn before me this 20th day of December 2000.

Vicky D. Schroer

Notary Public, State of California

VICKY D. SCHROER
Commission # 1224251
Notary Public - California
Santa Clara County
My Comm. Expires Jun 12, 2003

ATTACHMENT A

Proposed Changes to Operating Licenses and Technical Specifications for Dresden Nuclear Power Station, Units 2 and 3

DESCRIPTION AND SUMMARY SAFETY ANALYSIS

FOR PROPOSED CHANGES

ATTACHMENT A

Proposed Changes to Operating Licenses and Technical Specifications for Dresden Nuclear Power Station, Units 2 and 3

DESCRIPTION AND SUMMARY SAFETY ANALYSIS FOR PROPOSED CHANGES

A. SUMMARY OF PROPOSED CHANGES

Pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," Commonwealth Edison (ComEd) Company is requesting changes to the license and various Technical Specifications (TS) for Dresden Nuclear Power Station (DNPS), Units 2 and 3. The requested changes support an extended power uprate (EPU) for the DNPS units.

DNPS is a dual-unit site. Each unit is a General Electric (GE) Boiling Water Reactor (BWR)/3 with a Mark I containment. Because of the significant economic advantages of operating at higher power levels, ComEd is proposing permanent changes to the operating licenses to enable the DNPS units to be operated at levels up to 17 percent above the current rated power level of 2527 megawatts thermal (MWt). This increase corresponds to an uprated power level of 2957 MWt.

The analyses and evaluations supporting the proposed changes directly related to power uprate were completed using the guidelines in GE Topical Report NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate" (Reference I.1). Certain issues are evaluated generically and have been submitted to the NRC in GE Topical Report NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate" (Reference I.2). The NRC has approved both of these topical reports, in References I.3 and I.4, respectively.

The planned approach to achieving the higher power level consists of an increase in the core thermal power with a more uniform power distribution and reactor operation primarily along the Maximum Extended Load Line Limit Analysis (MELLLA) rod/flow control lines. The use of the MELLLA domain allows increased thermal power without an increase in core flow. The increased core thermal power will create increased steam flow and require a corresponding increase in the feedwater system flow, which will be achieved by operation of the third feedwater pump and the fourth condensate pump. DNPS is also proposing to implement the Average Power Range Monitor (APRM) / Rod Block Monitor (RBM) TS (ARTS) power and flow dependent limits to increase plant operational flexibility by updating the fuel thermal limit requirements. This application of ARTS is considered a partial application, as discussed in Section 9.2.1 of Attachment E, since these units are not implementing the hardware changes that are usually installed to the RBM system. The maximum allowable core flow rate does not change as a result of power uprate. In addition, uprated operation will not involve increasing reactor pressure vessel (RPV) dome pressure because the DNPS units have sufficient pressure control and turbine flow capabilities to control the inlet pressure conditions at the turbine. However, to maintain the GE standard turbine flow margin of three percent, modifications will be made to the high-pressure turbine. Attachment G describes the planned hardware modifications that will maintain adequate performance margins.

The proposed licensed power level of 2957 MWt is used as the basis for the Power Uprate Safety Analysis Report (PUSAR), provided in Attachment E, which supports the proposed changes. Attachment E demonstrates that DNPS can safely operate at the proposed licensed power level of 2957 MWt. The proposed licensed power level of 2957 MWt was chosen based on the following considerations. First, feasibility studies showed that a power level of at least 2898 MWt was required to produce an expected output of 912 megawatts electric (MWe), which is the current limitation on the output of the main generator. Second, operation at a power level somewhat greater than 2898 MWt may be required to achieve the 912 MWe output capability of the main generator because the effects of plant efficiencies when operating at the uprated power level can not be fully known prior to implementation. DNPS expects to operate the Unit 2 and 3 reactors at the power level required to achieve an electrical output of 912 MWe. This power level will vary with the conditions that effect plant thermal efficiency. Finally, future economic conditions may allow upgrade of the main generator and other related modifications to allow a further increase in electric output to take advantage of the proposed power level of 2957 MWt.

DNPS has submitted a TS amendment request (Reference I.6) for conversion from the Current TS (CTS) to the Improved Technical Specifications (ITS). In anticipation of approval of that request, this request for amendment is based on the format of the ITS. In addition, the affected sections of the CTS are noted.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

B.1. Operating License Maximum Power Level

Condition 2.C(1) of the current operating license for DNPS Unit 2 states that "The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2527 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein."

Condition 3.A of the current operating license for DNPS Unit 3 states that "Commonwealth Edison is authorized to operate the facility at steady state power levels not in excess of 2527 megawatts (thermal), except that Commonwealth Edison shall not operate the facility at power levels in excess of five (5) megawatts (thermal), until satisfactory completion of modifications and final testing of the station output transformer, the auto-depressurization interlock, and the feedwater system, as described in Commonwealth Edison's telegram; dated February 26, 1971, have been verified in writing by the Commission."

B.2. Operating License Condition on Containment Overpressure

DNPS Unit 2 has an operating license condition associated with TS Amendment 157 and DNPS Unit 3 has an operating license condition associated with TS Amendment 152 that states, "The license is amended to authorize changing the UFSAR to allow credit for containment overpressure as detailed below, to assure adequate Net Positive Suction Head is available for low pressure Emergency Core Cooling System pumps following a design basis accident."

Time (seconds)	Containment Pressure (PSIG)
0-240	9.5
240-480	2.9
480-6000	1.9
6000-accident end	2.5

B.3. TS Definition of Rated Thermal Power

ITS Section 1.1, "Definitions," defines Rated Thermal Power (RTP) as follows. "RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2527 MWt." RTP is also defined in CTS Section 1.0, "Definitions."

B.4. TS Definition of Fuel Design Limiting Ratio for Centerline Melt

ITS Section 1.1 states that the Fuel Design Limiting Ratio for Centerline Melt (FDLRC) shall be 1.2 times the Linear Heat Generation Rate (LHGR) existing at a given location divided by the product of the transient LHGR (TLHGR) and the fraction of RTP. FDLRC is also defined in CTS Section 1.0.

B.5. TS Section 3.2.4, "Average Power Range Monitor Gain and Setpoint"

ITS Section 3.2.4 requires that when thermal power is ≥ 25%, FDLRC be less than or equal to 1.0 or that each required APRM Flow Biased Neutron Flux - High Function Allowable Value be modified by 1/FDLRC or that each required APRM gain be adjusted such that the APRM readings are ≥ 100% times the Fraction of RTP (FRTP) times the FDLRC. CTS Section 3.11.B, "Transient Linear Heat Generation Rate," specifies the same requirement.

B.6. TS Section 3.3.1.1, "Reactor Protection System Instrumentation"

Several changes to the Reactor Protection System (RPS) Instrumentation TS are proposed. These include changes to Surveillance Requirements (SRs), Limiting Conditions for Operation (LCO), specified conditions, allowable values and action statements.

TS SR 3.3.1.1.2

ITS SR 3.3.1.1.2 requires verification that the absolute difference between the APRM channels and the calculated power is ≤ 2% RTP plus any gain adjustment required by LCO 3.2.4 while operating at ≥ 25% RTP. This requirement is also identified in CTS Table 4.1.A-1, "Reactor Protection System Instrumentation Surveillance Requirements."

TS SR 3.3.1.1.14

ITS SR 3.3.1.1.14 requires verification that the Turbine Stop Valve (TSV) - Closure and Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when thermal power is ≥ 45% RTP. This requirement is also identified in CTS Table 4.1.A-1.

TS Table 3.3.1.1-1 Function 2.b

ITS Table 3.3.1.1-1, "Reactor Protection System Instrumentation," Function 2.b identifies the allowable values for the APRM Flow Biased Neutron Flux - High Function. For two-loop operation, the allowable value is ≤ 0.58 W + 63.5% RTP and ≤ 122% RTP. For single-loop operation, the allowable value is ≤ 0.58 W + 59.2% RTP and ≤ 118.5% RTP. A similar requirement is specified in CTS Table 2.2.A-1, "Reactor Protection System Instrumentation Setpoints."

TS Table 3.3.1.1-1 Function 4

ITS Table 3.3.1.1-1, Function 4 identifies the allowable value for the Reactor Vessel Water Level – Low Function. The allowable value is ≥ 10.24 inches. A similar requirement is specified in CTS Table 2.2.A-1.

TS Table 3.3.1.1-1 Function 8

ITS Table 3.3.1.1-1 Function 8 specifies that the TSV – Closure Function is required to be operable when reactor power is ≥ 45% RTP. This requirement is also specified in CTS Table 3.1.A-1, "Reactor Protection System Instrumentation."

TS Table 3.3.1.1-1 Function 9

ITS Table 3.3.1.1-1 Function 9 specifies that the TCV Fast Closure, Trip Oil Pressure – Low Function is required to be operable when reactor power is ≥ 45% RTP. This requirement is also specified in CTS Table 3.1.A-1.

TS Table 3.3.1.1-1 Function 10

ITS Table 3.3.1.1-1 Function 10 specifies that the allowable value for the Turbine Condenser Vacuum – Low scram function be ≥ 21.15 inches HG vacuum. A similar requirement is also identified in CTS Table 2.2.A-1.

TS Section 3.3.1.1 Required Action E.1

ITS Section 3.3.1.1 Action E.1 requires THERMAL POWER to be reduced to < 45% RTP as required by Action D.1 and referenced in Table 3.3.1.1-1. This requirement is also identified in CTS Table 3.1.A-1.

B.7. TS Section 3.3.5.2, "Isolation Condenser System Instrumentation"

ITS SR 3.3.5.2.3 requires the performance of a channel calibration for the time delay portion of the Isolation Condenser (IC) channels. The requirement states that the allowable value shall be ≤ 17 seconds. This requirement is not included in the CTS although the 17 second time delay is discussed in CTS Bases Section 3/4.5.D.

B.8. TS Section 3.3.6.1, "Primary Containment Isolation Instrumentation"

Several changes to the Primary Containment Isolation Instrumentation TS are proposed.

TS Table 3.3.6.1-1 Function 1.d

ITS Table 3.3.6.1-1 provides a listing of the required Primary Containment Isolation Instrumentation. Item 1.d describes the requirements and allowable values for the main steam line (MSL) isolation function on Main Steam Line Flow - High. The allowable value for Unit 2 is ≤ 160.5 psid and the allowable value for Unit 3 is ≤ 117.1 psid. This requirement is also identified in CTS Table 3.2.A-1, “Isolation Actuation Instrumentation,” but is expressed in percent of rated steam flow.

TS Table 3.3.6.1-1 Function 2.a

ITS Table 3.3.6.1-1 provides a listing of the required Primary Containment Isolation Instrumentation. Item 2.a describes the requirements and allowable values for the primary containment isolation function on Reactor Vessel Water Level – Low. The allowable value is ≥ 10.24 inches. A similar requirement is specified in CTS Table 3.2.A-1.

TS Table 3.3.6.1-1 Function 5.b

ITS Table 3.3.6.1-1 provides a listing of the required Primary Containment Isolation Instrumentation. Item 5.b describes the requirements and allowable values for the Reactor Water Cleanup (RWCU) System isolation function on Reactor Vessel Water Level – Low. The allowable value is ≥ 10.24 inches. A similar requirement is specified in CTS Table 3.2.A-1.

TS Table 3.3.6.1-1 Function 6.b

ITS Table 3.3.6.1-1 provides a listing of the required Primary Containment Isolation Instrumentation. Item 6.b describes the requirements and allowable values for Reactor Vessel Water Level – Low function for shutdown cooling isolation system isolation. The allowable value is ≥ 10.24 inches. A similar requirement is specified in CTS Table 3.2.A-1.

B.9. TS Section 3.4.3, "Safety and Relief Valves"

ITS Section 3.4.3 requires the safety function of eight safety valves to be operable. SR 3.4.3.1 specifies the safety function lift setpoints of the safety valves. A similar requirement for DNPS Unit 3 is identified in CTS Section 3.6.E, "Safety Valves." However, until approval of the TS amendment request identified in Reference I.5 is obtained, nine safety valves are required to be operable for Unit 2.

B.10. TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program"

ITS Section 5.5.12 states that the peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 48 psig. This requirement is also identified in CTS Bases Section B 3/4.7.A, "Primary Containment Integrity."

B.11. TS Section 5.6.5, "Core Operating Limits Report"

ITS Section 5.6.5.a, Item 4 specifies that the core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the Core Operating Limits Report (COLR) including the LHGR and TLHGR limit for Specification 3.2.4. A requirement for the COLR exists in CTS Section 6.9.A.6.a, "Core Operating Limits Report," although there is no requirement to include the LHGR or TLHGR.

C. BASES FOR THE CURRENT REQUIREMENTS

C.1. Operating License Maximum Power Level

The current operating license and the affected TS sections are based on a RTP of 2527 MWt. The supporting transient and accident analyses justifying operation are also based on this RTP with appropriate margins added, in accordance with regulatory guidance. Limits placed on RTP, Reactor Coolant System (RCS) pressure, RCS temperature and flow ensure that the initial conditions will be met for each of the transients analyzed.

C.2. Operating License Condition on Containment Overpressure

To ensure that there is adequate net positive suction head (NPSH) to support the operation of the emergency core cooling system (ECCS) pumps during design basis accident (DBA) conditions, the analyses take credit for containment overpressure. This allowance was approved in TS Amendment 157 for DNPS Unit 2 and TS Amendment 152 for DNPS Unit 3.

C.3. TS Definition of Rated Thermal Power

The current operating licenses and the affected TS sections are based on a RTP of 2527 MWt. The supporting accident and transient analyses justifying operation were based on this power level with appropriate margin added in accordance with regulatory guidance.

C.4. TS Definition of Fuel Design Limiting Ratio for Centerline Melt

The condition of excessive power peaking is determined by FDLRC. When FDLRC is greater than 1.0, excessive power peaking exists. Maintaining FDLRC less than or equal to 1.0 ensures that the fuel does not experience centerline melt and protects against fuel cladding 1% plastic strain during Anticipated Operational Occurrences (AOOs) beginning at any power level and terminating at ≤ 122% RTP which corresponds to the APRM Fixed Neutron Flux - High allowable value.

C.5. TS Section 3.2.4, "Average Power Range Monitor Gain and Setpoint"

This LCO is provided to require the APRM gain or APRM Flow Biased Neutron Flux - High Function Allowable Value to be adjusted by a ratio defined by FDLRC when operating under conditions of excessive power peaking. This adjustment is necessary to maintain acceptable margin to the fuel cladding integrity safety limit and the fuel cladding 1% plastic strain limit. When the FDLRC is greater than 1.0, excessive power peaking exists and the APRM Flow Biased Neutron Flux - High Function allowable value must be adjusted to ensure that the TLHGR limit is not violated for any power distribution. To maintain margins similar to those at RTP conditions, the APRM flow biased allowable value is decreased by 1/FDLRC. As an alternative, the APRM gain can be increased by FDLRC. Increasing the APRM gain raises the initial APRM reading closer to the flow biased allowable value such that a scram would be received at the same point in a transient as if the allowable value had been reduced. Thus, providing the same degree of protection as reducing the APRM Flow Biased Neutron Flux - High Function Allowable Value by 1/FDLRC.

C.6. TS Section 3.3.1.1, "Reactor Protection System Instrumentation"

TS SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. TS LCO 3.2.4, "APRM Gain and Setpoint," allows the APRMs to be reading greater than actual thermal power to compensate for localized power peaking. SR 3.3.1.1.2 verifies that the absolute difference between the APRM channels and the calculated power is ≤ 2% plus any gain adjustment required by LCO 3.2.4.

TS SR 3.3.1.1.14

Since the TSV Closure and TCV Fast Closure, Trip Oil Pressure – Low Functions are capable of being bypassed when reactor power is sufficiently low, this SR ensures that these scram functions will not be bypassed when they may be needed to mitigate a Turbine/Generator (T/G) trip. The associated analyses are based on a reactor power of 45% or approximately 1137 MWt.

TS Table 3.3.1.1-1 Function 2.b

TS Table 3.3.1.1-1 Function 2.b is the Flow Biased Neutron Flux – High setpoint for the APRMs. The purpose of the APRMs is to generate a reactor trip signal on high neutron flux to prevent fuel damage or excessive RCS pressure. During operation, the neutron flux level varies with recirculation drive flow. At lower core flows, this setpoint is reduced as core flow is reduced but is clamped at an upper limit that is equivalent to the APRM Fixed Neutron – High Function allowable value. Because of a lower scram trip setpoint, the APRM Flow Biased Neutron Flux – High Function will initiate a scram before the clamped allowable value is reached during any transient event that occurs at a reduced recirculation flow.

TS Table 3.3.1.1-1 Function 4

TS Table 3.3.1.1-1 Function 4 identifies the instrumentation requirements for the Reactor Vessel Water Level – Low Function including the allowable value. A low RPV water level indicates that the capability to cool the fuel may be threatened. Should the RPV water level decrease too far, fuel damage could result. Therefore, a reactor scram is initiated at a low water level to substantially reduce the heat generated in the fuel from fission. The Reactor Vessel Water Level – Low allowable value is selected to ensure that during normal operation, the steam separator skirts are not uncovered to protect available recirculation pump NPSH from significant steam ingestion.

TS Table 3.3.1.1-1 Function 8

TS Table 3.3.1.1-1 Function 8 identifies the instrumentation requirements for the TSV – Closure Function including the operating conditions when the function is required to be operable. This function is required to be enabled when RTP is

≥ 45%, which corresponds to a reactor power level of approximately 1137 MWt. This item is identified in the table since this trip is capable of being bypassed at low power levels when the scram function is not needed to mitigate a T/G trip.

TS Table 3.3.1.1-1 Function 9

TS Table 3.3.1.1-1 Function 9 identifies the instrumentation requirements for the TCV Fast Closure, Trip Oil Pressure – Low Function including the operating conditions when the function is required to be operable. This function is required to be enabled when RTP is ≥ 45%, which corresponds to a reactor power level of approximately 1137 MWt. This item is identified in the table since this trip is capable of being bypassed at low power levels when the scram function is not needed to mitigate a T/G trip.

TS Table 3.3.1.1-1 Function 10

The Turbine Condenser Vacuum – Low Function is provided to shut down the reactor and reduce the energy input to the main condenser to help prevent overpressurization of the main condenser in the event of a loss of the main condenser vacuum. The Turbine Condenser Vacuum – Low Function is the primary scram signal for the loss of condenser vacuum event. For this event, the reactor scram reduces the amount of energy required to be absorbed by the main condenser. It also helps to ensure the MCPR safety limit is not exceeded by reducing the core energy prior to the fast closure of the turbine stop valves. This function helps maintain the main condenser as a heat sink during this event.

TS Section 3.3.1.1 Required Action E.1

If an associated RPS channel is not restored to operable status or placed in trip within the allowed completion time specified in Required Action E.1, the plant must be placed in a mode or other specified condition in which the LCO does not apply. This LCO is not applicable when reactor power is < 45% RTP.

C.7. TS Section 3.3.5.2, "Isolation Condenser System Instrumentation"

The purpose of the IC system instrumentation is to initiate actions to ensure adequate core cooling when the reactor vessel is isolated from its primary heat sink, the main condenser. The reactor vessel high-pressure initiation time delay is provided to avoid spurious unnecessary actuations.

C.8. TS Section 3.3.6.1, "Primary Containment Isolation Instrumentation"

TS Table 3.3.6.1-1 Function 1.d

Main Steam Line Flow – High is provided to detect a break of the MSL and to initiate closure of the Main Steam Isolation Valves (MSIVs). If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow – High Function is directly assumed in the analysis of the Main Steam Line Break (MSLB). The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and offsite doses do not exceed the 10 CFR 100, "Reactor Site Criteria," limits.

TS Table 3.3.6.1-1 Function 2.a

Primary containment isolation on Reactor Vessel Water Level – Low is provided to isolate the valves whose penetrations communicate with the primary containment to limit the release of fission products when the RPV water level indicates that the capability to cool the fuel may be threatened. The isolation of the primary containment on low RPV level supports actions to ensure that the offsite dose limits of 10 CFR 100 are not exceeded. This isolation function is implicitly assumed in the Updated Final Safety Analysis Report (UFSAR) analysis as these leakage paths are assumed to be isolated after a Loss of Coolant Accident (LOCA). The allowable value associated with this function was chosen to be the same as the RPS Reactor Vessel Water Level – Low scram allowable value, since isolation of these valves is not critical to orderly plant shutdown.

TS Table 3.3.6.1-1 Function 5.b

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the RPV occurs to isolate the potential sources of a break. The isolation of the RWCU system on low RPV water level supports actions to ensure that the fuel peak cladding temperature remains below the limits specified in 10 CFR 50.46. The RWCU isolation function is not directly assumed in the UFSAR safety analyses because the RWCU system line break is bounded by breaks of larger systems.

TS Table 3.3.6.1-1 Function 6.b

This function is associated with the isolation of the shutdown cooling system and is only required to be operable in modes 3, 4, and 5. Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the RPV occurs to isolate the potential sources of a break. The isolation of the shutdown cooling system is not directly assumed in the UFSAR safety analyses because the shutdown cooling system break is bounded by breaks of the recirculation and main steam lines.

C.9. TS Section 3.4.3, "Safety and Relief Valves"

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requires that the RPV be protected from overpressurization during upset conditions by self-actuated safety valves. The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all MSIVs, followed by a reactor scram on high neutron flux with a failure of the direct scram associated with MSIV position. For the purpose of the analysis of this event, eight safety valves are assumed to operate in the safety mode. The results of the analysis demonstrate that the design safety valve capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure. Operation with fewer valves operable than specified, or with setpoints outside the ASME limits, could result in a more severe reactor response to a transient than predicted. This LCO helps to ensure that the acceptance limit is met during the design basis event.

C.10. TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program"

The maximum design pressure for the containment is 62 psig. The safety analysis associated with the postulated design basis LOCA predicts a peak containment pressure of 47 psig. Containment pressure testing is performed at 48 psig to ensure leakage rates are within the criteria established to ensure offsite doses do not exceed the limits of 10 CFR 100.

C.11. TS Section 5.6.5, "Core Operating Limits Report"

Cycle specific parameters, previously located in the TS, have been relocated to the COLR. To support the determination of the FDLRC as required by TS 3.2.4, the LHGR and the TLHGR limits are required to be submitted in the COLR.

D. NEED FOR REVISION OF THE REQUIREMENTS

D.1. Operating License Maximum Power Level

The proposed changes allow an increase in licensed core thermal power from 2527 MWt to 2957 MWt and provide the flexibility to increase the potential electrical output of DNPS, Units 2 and 3. This power uprate will provide a net increase of approximately 206 MWe in generation to serve commercial and domestic loads on the electrical grid.

D.2. Operating License Condition on Containment Overpressure

The analysis associated with the postulated LOCA at increased power levels results in an increase in suppression pool water temperature. Because of the increase in water temperature, the need for additional credit for containment overpressure to maintain adequate NPSH for the ECCS pumps has been identified.

D.3. TS Definition of Rated Thermal Power

The proposed changes allow an increase in licensed core thermal power from 2527 MWt to 2957 MWt and provide the flexibility to increase the potential electrical output of the DNPS, Units 2 and 3. This change is needed to support the change identified in section D.1 of this attachment.

D.4. TS Definition of Fuel Design Limiting Ratio for Centerline Melt

The ARTS power and flow dependent limits provide additional thermal limit restrictions. These allow the removal of the requirement to modify the APRM gain and setpoint based on the FDLRC as discussed in Sections 1.4.1 and 9.2 of Attachment E. The elimination of this requirement also results in the elimination of the requirement to perform the FDLRC calculations.

D.5. TS Section 3.2.4, "Average Power Range Monitor Gain and Setpoint"

With the implementation of the ARTS power and flow dependent limits, the additional restrictions that are imposed facilitate the removal of the requirement to modify the APRM gain and setpoint based on the FDLRC as discussed in Sections 1.4.1 and 9.2 of Attachment E.

D.6. TS Section 3.3.1.1, "Reactor Protection System Instrumentation"

<u>TS SR 3.3.1.1.2</u>

Since the proposed changes remove TS Section 3.2.4, TS SR 3.3.1.1.2 must be modified to remove the reference to TS Section 3.2.4.

<u>TS SR 3.3.1.1.14</u>

The proposed changes revise the percent RTP at which the TSV – Closure and the TCV Fast Closure, Trip Oil Pressure - Low Functions are verified not to be bypassed. The actual power level at which these trips are required to be operable remains the same. 45% of pre-uprate RTP is essentially the same value as 38.5% of post-uprate RTP as described in Section F.6 of this attachment.

<u>TS Table 3.3.1.1-1 Function 2.b</u>

The proposed changes revise the allowable values for the APRM Flow Biased Neutron Flux – High Function to be consistent with the ELTR (References I.1 and

I.2) and the MELLLA. New analytical limits for the flow biased APRM scrams for two-loop operation and single-loop operation have been developed for uprated power conditions.

TS Table 3.3.1.1-1 Function 4

The loss of feedwater (LOFW) transient was reanalyzed under EPU conditions. Due to increased core heat generation, the RPV water level decreases more rapidly in this transient. A plant modification is being installed to add a recirculation pump runback function to reduce the effects of this water level decrease. Lowering the reactor vessel low water level scram setpoint will increase the potential for recovery before reaching the scram setpoint and thus prevent unnecessary challenges to safety systems and provide additional time for operator action.

TS Table 3.3.1.1-1 Function 8

The proposed changes revise the percent RTP at which the TSV - Closure Function is verified not to be bypassed. The new percent RTP is required to maintain the existing thermal power level at which the function is currently verified. 45% of pre-uprate RTP is essentially the same value as 38.5% of post-uprate RTP as described in Section F.6 of this attachment.

TS Table 3.3.1.1-1 Function 9

The proposed changes revise the percent RTP at which the TCV Fast Closure, Trip Oil Pressure - Low Function is currently verified not to be bypassed. The new percent RTP is required to maintain the existing thermal power level at which the function is verified. 45% of pre-uprate RTP is essentially the same value as 38.5% of post-uprate RTP as described in Section F.6 of this attachment.

TS Table 3.3.1.1-1 Function 10

With the increased heat input due to EPU, the backpressure in the condenser will rise. The plant has an alarm for condenser low vacuum at a nominal value of 24.5 inches of Hg with a scram allowable value of 21.15 inches of Hg. In conditions of high ambient temperature, the condenser backpressure could potentially exceed the alarm setpoint. To avoid this alarm during normal operations, the alarm setpoint is being changed. To maintain adequate margin between the alarm and the scram, the scram allowable value is being changed to 21.4 inches Hg. The analytical limit for the function remains unchanged.

TS Section 3.3.1.1 Required Action E.1

TS Action E.1 requires that thermal power be reduced to < 45% RTP in the event Condition E is entered. The proposed change revises the TS Action to reduce RTP to < 38.5% of the proposed RTP in the event TS Section 3.3.1.1 Condition E was entered to maintain the actual value of reactor power consistent with the pre-uprate value.

D.7. TS Section 3.3.5.2, "Isolation Condenser System Instrumentation"

To ensure compliance with the transient analysis, the IC automatic initiation function time delay must be reduced from 17 seconds to 15 seconds. The analysis of the LOFW transient event was performed using a reactor vessel high-pressure initiation time delay of 15 seconds.

D.8. TS Section 3.3.6.1, "Primary Containment Isolation Instrumentation"

TS Table 3.3.6.1-1 Function 1.d

The existing analytical limits for the Main Steam Line Flow – High Function correspond to 120% of rated steam flow. To provide for increased protection against unnecessary MSL isolation during valve testing, the proposed analytical limits will be increased to the equivalent of 140% of rated steam flow at EPU conditions for Unit 3 and 125% of rated steam flow at EPU conditions for Unit 2. The lower value for Unit 2 is due to limitations associated with the Unit 2 flow restrictors.

TS Table 3.3.6.1-1 Function 2.a

This change is associated with TS Table 3.3.1.1-1 Function 4, of Item 6. This item lowers the allowable value of the Reactor Vessel Water Level – Low RPS scram function. To maintain the isolation function at the same level, the allowable value for TS Table 3.3.6.1-1 Function 2.a must also be revised.

TS Table 3.3.6.1-1 Function 5.b

This change is associated with TS Table 3.3.1.1-1 Function 4, of Item 6. This item lowers the allowable value of the Reactor Vessel Water Level – Low RPS scram function. To maintain the isolation function at the same level, the allowable value for TS Table 3.3.6.1-1 Function 2.a must also be revised.

TS Table 3.3.6.1-1 Function 6.b

This change is associated with TS Table 3.3.1.1-1 Function 4, of Item 6. This item lowers the allowable value of the Reactor Vessel Water Level – Low RPS scram function. To maintain the isolation function at the same level, the allowable value for TS Table 3.3.6.1-1 Function 2.a must also be revised.

D.9. TS Section 3.4.3, "Safety and Relief Valves"

The existing requirement to ensure that the safety functions of eight safety valves are operable is necessary to support the analysis associated with MSIV closure followed by a reactor scram on high neutron flux. This transient assumes the failure of the direct scram associated with MSIV closure. The existing TS does not credit the use of the Target Rock safety/relief valve and thus the safety function lift setpoint is not specified in SR 3.4.3.1. Although the transient conservatively assumes the failure of this valve since it has the highest capacity, to credit the use of this valve in the case that one of the other safety valves is out of service, it is necessary to include the lift setpoint in the TS SR.

D.10. TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program"

The analysis of the postulated DBA-LOCA using a more detailed model has identified a lower predicted peak containment pressure compared to the pressure at which the containment is currently tested as identified in the TS. Revising the TS to match the results of the analysis will result in a reduction of burden without affecting the safety analysis.

D.11. TS Section 5.6.5, "Core Operating Limits Report"

The proposed changes remove TS Section 3.2.4 as part of the implementation of the ARTS power and flow dependent limits as described in Item 5 of this attachment. With this change, the inclusion of LHGR and the TLHGR in the COLR for Specification 3.2.4 is no longer necessary.

E. DESCRIPTION OF THE PROPOSED CHANGES

Unless otherwise stated, the affected TS sections are the same for Unit 2 and Unit 3.

E.1. Operating License Maximum Power Level

Condition 2.C(1) of the current operating license for DNPS Unit 2 is revised to state that, "The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2957 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein."

Condition 3.A of the current operating license for DNPS Unit 3 is revised to state that, "The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2957 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein."

E.2. Operating License Condition on Containment Overpressure

The allowance for containment overpressure in the license conditions is revised to state, "The license is amended to authorize changing the UFSAR to allow credit for containment overpressure as detailed below, to assure adequate Net Positive Suction Head is available for low pressure Emergency Core Cooling System pumps following a design basis accident."

Time (seconds)	Containment Pressure (PSIG)
0-290	9.5
290-5,000	4.8
5,000-30,000	4.25

E.3. TS Definition of Rated Thermal Power

Section 1.1, "Definitions," RTP is revised to reflect the increase from 2527 MWt to 2957 MWt.

E.4. TS Definition of Fuel Design Limiting Ratio for Centerline Melt

The definition of FDLRC in Section 1.1, "Definitions," is deleted.

E.5. TS Section 3.2.4, "Average Power Range Monitor Gain and Setpoint"

TS Section 3.2.4 is deleted.

E.6. TS Section 3.3.1.1, "Reactor Protection System Instrumentation"

TS SR 3.3.1.1.2

The reference to TS Section 3.2.4 is removed so that SR 3.3.1.1.2 states, "Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP."

TS SR 3.3.1.1.14

The thermal power applicability is changed from ≥ 45% to ≥ 38.5% so that SR 3.3.1.1.14 states, "Verify Turbine Stop Valve – Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure – Low Functions are not bypassed when THERMAL POWER is ≥ 38.5%."

TS Table 3.3.1.1-1 Function 2.b

The allowable value for the APRM Flow Biased Neutron Flux – High Function is changed to 0.56 W + 67.4% and ≤ 122% for two-loop operation and 0.56 W + 63.2% and ≤ 118.5% for single-loop operation as identified in note (b) of Table 3.3.1.1-1.

TS Table 3.3.1.1-1 Function 4

The allowable value for the Reactor Vessel Water Level – Low function is reduced by approximately 8 inches from ≥ 10.24 inches to ≥ 2.65 inches.

TS Table 3.3.1.1-1 Function 8

The value in the column labeled "Applicable Modes or Other Specified Conditions" is changed from ≥ 45% to ≥ 38.5%.

TS Table 3.3.1.1-1 Function 9

The value in the column labeled "Applicable Modes or Other Specified Conditions" is changed from ≥ 45% to ≥ 38.5%.

TS Table 3.3.1.1-1 Function 10

The allowable value for the Turbine Condenser Vacuum – Low function is changed from ≥ 21.15 inches Hg vacuum to ≥ 21.4 inches Hg vacuum.

TS Section 3.3.1.1 Required Action E.1

The reference to the thermal power level in Required Action E.1 is changed from < 45% to < 38.5% so that Action E.1 states, "Reduce THERMAL POWER to < 38.5% RTP."

E.7. TS Section 3.3.5.2, "Isolation Condenser System Instrumentation"

The allowable value for the time delay in TS SR 3.3.5.2.3 is changed from ≤ 17 seconds to ≤ 15 seconds.

E.8. TS Section 3.3.6.1, "Primary Containment Isolation Instrumentation"

TS Table 3.3.6.1-1 Function 1.d

The allowable value is changed from ≤ 160.5 psid for Unit 2 and ≤ 117.1 psid for Unit 3 to ≤ 259.2 psid for Unit 2 and ≤ 252.6 psid for Unit 3. These values correspond to 125% of rated steam flow for Unit 2 and 140% of rated steam flow for Unit 3.

TS Table 3.3.6.1-1 Function 2.a

The allowable value for the Reactor Vessel Water Level – Low function is reduced by approximately 8 inches from ≥ 10.24 inches to ≥ 2.65 inches.

TS Table 3.3.6.1-1 Function 5.b

The allowable value for the Reactor Vessel Water Level – Low function is reduced by approximately 8 inches from ≥ 10.24 inches to ≥ 2.65 inches.

TS Table 3.3.6.1-1 Function 6.b

The allowable value for the Reactor Vessel Water Level – Low function is reduced by approximately 8 inches from ≥ 10.24 inches to ≥ 2.65 inches.

E.9. TS Section 3.4.3, "Safety and Relief Valves"

SR 3.4.3.1 is revised to include the safety function lift setpoint for the Target Rock safety valve. The setpoint for the safety valve is 1135 ± 11.3 psig.

E.10. TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program"

TS 5.5.12.b is revised to reflect a peak calculated primary containment internal pressure for the design basis LOCA, P_a, of 43.9 psig.

E.11. TS Section 5.6.5, "Core Operating Limits Report"

TS Section 5.6.5.a.4 is deleted.

F. SUMMARY SAFETY ANALYSIS OF THE PROPOSED CHANGES

F.1. Operating License Maximum Power Level

The proposed changes increase the RTP from 2527 MWt to 2957 MWt. The detailed safety analyses for the proposed changes are contained in Attachment E. The analyses demonstrate that DNPS Units 2 and 3 can operate safely with the proposed 17 percent increase in maximum core thermal power with a corresponding 19 percent increase in steam flow from the RPV. The analyses also support the required increases of the flow, temperature, and pressure in the supporting systems and components.

DNPS, Units 2 and 3, are currently licensed for a 100 percent reactor power level of 2527 MWt and most of the current safety analyses are based on this value. However, the ECCS-LOCA and containment safety analyses are based on a power level of 1.02 times the licensed power level as required by Regulatory Guide 1.49, "Power Levels of Water-Cooled Nuclear Power Plants." The proposed uprated power level of 2957 MWt is approximately 17 percent greater than the currently licensed thermal power level. The EPU safety analyses are based on a power level of at least 1.02 times the EPU power level, except that some analyses are performed at 100% of uprated power, because the Regulatory Guide 1.49 two percent power factor is already accounted for in the analysis methods.

The analyses presented in Attachment E ensure that the power-dependent margin prescribed by Regulatory Guide 1.49 is maintained. For the safety analyses, NRC-accepted computer codes and calculational techniques are used to demonstrate compliance with the applicable regulatory acceptance criteria. Similarly, factors and margins specified by the application of design code rules is maintained, as are other margin-assuring acceptance criteria used to judge the acceptability of the plant. A list of the computer codes used for the EPU evaluations is provided in Attachment E, Table 1-3, "Computer Codes Used for EPU."

Effects on Plant Systems

Plant systems and components have been verified to be capable of performing their intended design functions at uprated power conditions, with some minor exceptions. Modifications to plant components necessary to support power uprate are identified in Attachment G. The review has concluded that operation at power uprate conditions will not affect the reliability of plant equipment.

Fuel Design Considerations

As discussed in Attachment E, Section 2, "Reactor Core and Fuel Performance," EPU increases the power density proportional to the power increase. However, this power density is still within the current operating power density range of most other BWRs. A representative equilibrium cycle core of GE14 fuel was used for the uprate evaluation. NRC approved core design methods were used to analyze core performance at EPU. The cycle specific reload core designs for operation at the uprated power level will take into account the above limits, to ensure acceptable differences between the licensing limits and their corresponding operating values.

At uprated conditions, all fuel and core design limits continue to be met by planned deployment of fuel enrichment and burnable poison management, control rod pattern and/or core flow adjustments.

Thermal-hydraulic design and operating limits ensure an acceptably low probability of boiling transition-induced fuel cladding failure occurring in the core, even for the most severe postulated operational transients. If needed, limits will be placed on fuel average planar linear heat generation rates to meet peak cladding temperature limits for the limiting LOCA.

EPU may result in a small change in fuel burnup, the amount of fuel to be used and isotopic concentrations of the radionuclides in the irradiated fuel relative to the current level of burnup. NRC approved limits for burnup on the fuel designs are not exceeded.

Capability of Makeup Water Sources

EPU with ARTS power and flow dependent limits does not result in an increase or decrease in the available water sources, nor does it change the selection of those assumed to function in the safety analyses. NRC approved methods were used for analyzing the performance of the ECCS during postulated loss of coolant accidents.

Design Basis Accidents

A review of DBAs was performed. DBAs are very low-probability hypothetical events whose characteristics and consequences are used in the design of the plant so that the plant can mitigate their consequences to within acceptable regulatory limits. For BWR licensing evaluations, capability is demonstrated for coping with the range of hypothetical pipe break sizes in the largest recirculation, steam, and feedwater lines, a postulated break in one of the ECCS lines, and the most limiting small lines. This break range bounds the full spectrum of large and small, high and low energy line breaks. The evaluation also accommodates a single active equipment failure in addition to the postulated LOCA coincident with a loss of offsite power (LOOP). Several of the most significant licensing assessments are made using these LOCA ground rules. These assessments are challenges to fuel, challenges to containment, and DBA radiological consequences.

- Challenges To Fuel

  The ECCS is described in UFSAR Section 6.3, "Emergency Core Cooling System." The ECCS performance evaluation, described in Section 4.3, "Emergency Core Cooling System Performance," of Attachment E, was conducted through application of 10 CFR 50 Appendix K, "ECCS Evaluation Models." This evaluation demonstrates the continued conformance to the acceptance criteria of 10 CFR 50.46. As mentioned above, a complete spectrum of pipe breaks is investigated from the largest recirculation line down to the most limiting small line break. The effect of the increased power level on the calculated peak cladding temperature (PCT) has been shown to be less than 10 degrees F as discussed in Section 4.3 of Attachment E. The increased PCT consequences for EPU with ARTS power and flow dependent limits remain within the fuel design limits and below the regulatory criteria. Therefore, the ECCS safety margin is not affected by EPU with ARTS power and flow dependent limits.

- Challenges to Containment

The containment analyses are described in UFSAR Section 6.2, "Containment Systems." The primary criteria of merit are the maximum containment pressure calculated during the course of the LOCA and maximum suppression pool temperature for long-term cooling in accordance with 10 CFR 50 Appendix A Criterion 38, "Containment Heat Removal."

Table 4-1, "DBA-LOCA Containment Performance Results," in Attachment E provides the results of the analyses of the plant containment responses to the most severe LOCAs. The effect of EPU on the peak values for containment pressure and temperature confirms the suitability of the plant for operation at EPU. Also, the effects of EPU on the conditions that affect the containment dynamic loads are determined, and the plant is judged satisfactory for EPU operation. The change in short-term containment response is negligible. Because there will be more residual heat with EPU, the containment long-term response increases slightly. However, containment pressures and temperatures remain below their design limits following any DBA, and thus, the containment and its cooling systems are judged to be satisfactory for EPU operation.

- Radiological Consequences

The UFSAR provides the radiological consequences for each DBA. The magnitude of the potential consequences is dependent upon the quantity of fission products released to the environment, the atmospheric dispersion factors and the dose exposure pathways. The atmospheric dispersion factors and the dose exposure pathways do not change. Therefore, the only factor that could influence the magnitude of the consequences is the quantity of activity released to the environment. This quantity is a product of the activity released from the core or reactor coolant and the transport mechanisms between the source region and the effluent release point. The transport mechanisms between the source region and the effluent release point are unchanged by EPU.

As discussed in Section 9.3, "Design Basis Accidents," of Attachment E, the events evaluated are the LOCA, the Main Steam Line Break Accident (MSLBA) outside containment, the Fuel Handling Accident (FHA), the CRDA, the Instrument Line Break (ILB) and the Offgas Treatment System Component Failure.

The EPU will not change the radiological consequences of a MSLBA outside containment, since the mass and energy releases following a MSLBA remain unaffected by uprate, and the activity released is based on primary coolant activity at TS levels, which is also unaffected by uprate.

The EPU will not change the radiological consequences of an ILB outside containment since the reactor coolant mass release used in the current analysis envelops the post-EPU conditions, and the activity released is based on primary coolant at TS levels which is unaffected by uprate.

The EPU will not change the radiological consequences of an Offgas Treatment System Component Failure since a conservative source term was used in the original analysis.

For the remaining DBAs, the primary parameter of importance is the activity released from the fuel. Because the mechanism of fuel failure is not influenced by EPU, the only parameter of importance is the actual inventory of fission products in the fuel rod. The only parameters affecting fuel inventory are the increase in thermal power, and to some extent, the cycle length.

The DBA which has historically been limiting from a radiological viewpoint is the LOCA, for which Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors" or its equivalent, has been applied. For this accident, it is assumed that 100% of the noble gases and 50% of the iodines in the core are released to the primary containment. These release fractions are not influenced by EPU or cycle length. As shown in Section 9.3, "Design Basis Accidents," of Attachment E, the LOCA dose consequences following uprate remain below regulatory guidelines. The EPU LOCA evaluation results include the 2% power uncertainty factor from Regulatory Guide 1.49.

The results of all radiological analyses remain below the 10 CFR 100 guideline values. Therefore, all radiological safety margins are maintained.

Transient Analyses

The effects of plant transients are evaluated in Section 9.1, "Reactor Transients," of Attachment E by investigating a number of disturbances of process variables and malfunctions or failures of equipment according to a scheme of postulating initiating events. These events are primarily evaluated against the Safety Limit Minimum Critical Power Ratio (SLMCPR). The SLMCPR is determined using NRC approved methods. The most limiting transient is slightly more severe when initiated from the uprated power level and results in a slightly larger change in MCPR than when initiated from the current power level. The result is less than a 0.03 change in MCPR. The Operating Limit MCPR is increased appropriately to assure that the SLMCPR is not challenged if any transient is initiated from the uprated power level. In addition, the limiting transients are analyzed for each specific fuel cycle. Licensing acceptance criteria are not exceeded. Therefore, the margin of safety is not affected by EPU.

Environmental Qualification

As discussed in Section 10.3, "Environmental Evaluation," of Attachment E, plant equipment and instrumentation has been evaluated against the criteria appropriate for uprate. Significant groups/types of the equipment have been justified for uprate by generic evaluations. In some cases, the qualification envelope did not change significantly due to uprate. A process has been developed to ensure qualification of the equipment whose current qualification does not already bound EPU conditions.

Fire Protection

A plant-specific evaluation assuming EPU conditions was performed to demonstrate safe shutdown capability in compliance with the requirements of 10 CFR 50 Appendix R, "Fire Protection Program For Nuclear Power Facilities Operating Prior To January 1, 1979." As discussed in Section 6.7.1, "10 CFR 50 Appendix R, Fire Event." of Attachment E, the results demonstrate EPU has no adverse impact on the ability to satisfy the requirements of Appendix R with respect to achieving and maintaining safe shutdown in the event of a fire. Minor procedure changes will be implemented to ensure the continued performance of the HPCI system during a safe shutdown scenario.

Instrumentation

The control and instrumentation signal ranges and analytical limits for setpoints are evaluated to establish the effects of the changes in various process parameters such as power, neutron flux, steam flow and feedwater flow. Analyses are performed to determine the need for setpoint changes for various functions such as MSL high flow isolation setpoints. In general, setpoints are changed only to maintain adequate operating margins between plant operating parameters and trip values, and only if satisfactory safety performance is demonstrated.

The instruments and controls that directly interact with or control the reactor are usually considered within the Nuclear Steam Supply System (NSSS). The NSSS process variables, instrument setpoints and Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," instrumentation that could be affected by EPU were evaluated. As part of EPU implementation, both the ComEd and GE setpoint methodologies were used to generate the allowable values and nominal trip setpoints related to the analytical limit changes.

TS instrument allowable values and/or setpoints are those sensed variables, which initiate protective actions. The determination of instrument allowable values and setpoints is based on plant operating experience and the conservative analytical limits used in specific licensing safety analyses. The settings are selected with sufficient margin to preclude inadvertent initiation of the protective action, while assuring that adequate operating margin is maintained between the system settings and the actual limits.

Increases in the core thermal power and steam flow affect some instrument setpoints, as described in Section 5.3, "Instrument Setpoints," of Attachment E. These setpoints are adjusted to maintain comparable differences between system settings and actual limits, and reviewed to assure that adequate operational flexibility and necessary safety functions are maintained at the extended uprated power level.

F.2. Operating License Condition on Containment Overpressure

EPU increases the reactor decay heat, which increases the heat input to the suppression pool in the event of a LOCA. This increased heat input could potentially increase the peak suppression pool water temperature and containment pressure during the post LOCA short-term and long-term low pressure core injection (LPCI) and core spray (CS) pump operation.

The ECCS NPSH requirements were evaluated for EPU conditions based on the pressure and temperature conditions determined by the containment analysis provided in Section 4.1.1," Containment Pressure and Temperature Response," of Attachment E, flow requirements based on the containment and LOCA analyses provided in Section 4.3 of Attachment E and flow losses, including suction strainer losses, determined using methodology previously reviewed by the NRC.

Calculations show that the available NPSH margins for the CS and LPCI pumps are not reduced during the short-term or long-term period following a DBA-LOCA. As with the original design analysis, the NPSH calculation does take credit for the wetwell airspace pressure during both short-term and long-term periods, as shown in Table 4-2, "NPSH Overpressure Credit," of Attachment E.

The credit taken for wetwell airspace pressure is adjusted for EPU conditions. This adjustment maintains the same (or greater) margin between the credited pressure profile and the analytical profile and the same (or greater) margin between the credited pressure profile and the pressure required for operation of each pump. For the EPU analysis, the credit taken during short-term and long-term periods is listed in Table 4-2 of Attachment E.

Short-term and long-term post-LOCA NPSH concerns are not applicable to the High Pressure Core Injection (HPCI) system. The available NPSH and required NPSH for the HPCI pump are not changed for EPU.

F.3. TS Definition of Rated Thermal Power

Revising the licensed RTP in Section 1.1 is associated with the increase in RTP described in Section F.1 of this attachment.

F.4. TS Definition of Fuel Design Limiting Ratio for Centerline Melt

Deleting the definition of FDLRC in TS Section 1.1 is associated with the implementation of the ARTS power and flow dependent limits. The definition of FDLRC is associated with the APRM gain and setpoint requirement of TS 3.2.4. The removal of this definition is associated with the deletion of TS 3.2.4 as described in Section F.5 of this attachment.

F.5. TS Section 3.2.4, "Average Power Range Monitor Gain and Setpoint"

The proposed change deletes the APRM gain and setpoint requirement. This requirement provides an operational restriction to ensure that the FDLRC does not exceed 1.0. This ensures that an acceptable margin to the fuel cladding integrity safety limit and the fuel cladding 1% strain limit is maintained. As discussed in Section 9.2, "Transient Analysis for ARTS Power and Flow Dependent Limits," of Attachment E, as a result of the implementation of the ARTS power and flow dependent limits, the operational restrictions associated with the APRM gain and setpoint adjustments to ensure FDLRC does not exceed 1.0 are bounded and can therefore be eliminated. This application of ARTS is a partial application. These units are not implementing the hardware changes that are usually installed to the RBM system. The hardware changes to the RBM system would typically provide the required protection for an off-rated RWE event. Therefore, off-rated RWE analyses were performed assuming the current RBM configuration with no rod blocks. The results of the off-rated RWE analyses showed that the generic K(P) and the plant specific MCPR(P) limits bound the results of the off-rated RWE event with no rod block. This analysis also supports the RBM operability power level ≥ 30% power. With the RBM inoperable below 30% power, the MCPR safety limit is protected by the MCPR(P) limits below Pbypass.

F.6. TS Section 3.3.1.1, "Reactor Protection System Instrumentation"

TS SR 3.3.1.1.2

The proposed changes remove the reference to the gain adjustment required by TS Section 3.2.4, as the APRM gain and setpoint requirements are superceded by the ARTS power and flow dependent limits related changes including the removal of TS Section 3.2.4. This change is a subset of the changes discussed in Section F.5 of this attachment.

TS SR 3.3.1.1.14

The TSV closure and TCV fast closure scrams can be bypassed when reactor power is sufficiently low and the scram function is not needed to mitigate a T/G trip. This power level, 38.5% RTP, is the analytical limit for determining the actual trip setpoint, which comes from the turbine first stage pressure (TFSP). The TFSP setpoint is chosen to allow operational margin so that scrams can be avoided, by transferring steam to the turbine bypass system during T/G trips at low power.

Based on the guidelines in Section F.4.2.3 of Reference I.1, the TSV Closure and TCV Fast Closure Scram Bypass analytical limits expressed as a percent of RTP are reduced by the ratio of the power increase such that the absolute power level at which the scram functions are required remains unchanged.

The existing RTP value for which the trip functions are verified not to be bypassed is 45% of 2527 MWt or approximately 1137 MWt. The uprated RTP value for which the trip functions are verified not to be bypassed is 38.5% of 2957 MWt or approximately 1138 MWt. The difference is negligible at approximately 0.1%. As a result, the new analytical limit does not change with respect to absolute thermal power and steam flow, and the setpoint does not change in terms of absolute power. Thus, there is no effect on the transient response. As a result, the same maneuvering range for plant startup is retained. The high-pressure turbine rotor modification will change the relationship between turbine first stage pressure and steam flow. Consequently, the scram bypass analytic limit in terms of measured pressure in psia must change to assure that the scram bypass occurs at or below the desired core thermal power and turbine steam flow point. However, the analytic limit as a percent of RTP is not changed by the rotor modification.

TS Table 3.3.1.1-1 Function 2.b

The proposed change revises the APRM flow biased scram equations for reactor recirculation two-loop and single-loop operation. The APRM Flow Biased Neutron Flux – High Function provides protection against transients where thermal power increases slowly, such as the recirculation loop flow controller failure event with increasing flow and the loss of feedwater heating event. This function also protects the fuel cladding integrity by ensuring that the MCPR safety limit is not exceeded. Because of a lower scram trip setpoint, the APRM Flow Biased Neutron Flux – High Function will initiate a scram before the clamped allowable value is reached during any transient event that occurs at a reduced recirculation flow. These changes are necessary to ensure consistent operation with the MELLLA power/flow map as discussed in Section 5.3.5, "Neutron Monitoring System," of Attachment E.

TS Table 3.3.1.1-1 Function 4

The proposed change lowers the allowable value for the Reactor Vessel Water Level – Low Function by 8 inches. The allowable value for the low water level signal is specified so that during normal operation, the seal skirts of the separators and dryers are covered. This is a requirement for plant operation and does not affect the licensing or safety basis of the plant. The allowable value is also specified so that the quantity of coolant following a low water level scram is sufficient for transients involving loss of all normal feedwater flow. Thus, the only transient that could be affected by lowering the scram level setpoint is the LOFW transient. This transient was evaluated to demonstrate that the setpoint change has no adverse effect on the reactor response. Since the LOFW is not a limiting MCPR event, the evaluation was performed primarily to demonstrate that there was no impact on the vessel inventory. In the LOFW event, the reactor water level decreases quickly causing a reactor scram on low water level. Following the scram, the reactor level continues to drop until it reaches the low-low level where the HPCI system will initiate to maintain the reactor water level and the IC system will provide core cooling. In addition, the reactor vessel low-low water level signal actuates closure of the MSIVs to limit the amount of inventory leaving the vessel. Lowering the low water level scram setpoint by 8 inches would delay the reactor scram for this event by a few seconds. However, since the setpoint for initiating HPCI at the low-low water level setpoint remains unchanged, there is no adverse impact on the ability of the system to maintain vessel inventory, and there is no impact on thermal margins. This is also discussed in Section 5.3.8, "Reactor Water Level Instruments," of Attachment E. Postulated LOCAs inside the containment are the most limiting in terms of peak clad temperature (PCT). This is because the postulated line break outside the containment is isolated before the reactor inventory loss out of the break can result in uncovering the core. Both large and small breaks were reviewed to determine the impact of lowering the analytical limit of the low water level scram by 8 inches. It was concluded that ECCS initiation and containment isolation will not be impacted because the time of scram will not change, since for these breaks, the high drywell pressure signal will occur before the low water level scram signal. Therefore, lowering the scram water level will not change the time of scram for any breaks inside containment and thus will not have a significant impact on ECCS initiation time or PCT.

TS Table 3.3.1.1-1 Function 8

This change is associated with the change in RTP for which the TSV - Closure Function is verified not to be bypassed and is described in Section F.6 in Subsection TS SR 3.3.1.1.14 of this attachment.

TS Table 3.3.1.1-1 Function 9

This change is associated with the change in RTP for which the TCV Fast Closure, Trip Oil Pressure - Low Function is verified not to be bypassed and is described in Section F.6 in Subsection TR SR 3.3.1.1.14 of this attachment.

TS Table 3.3.1.1-1 Function 10

This change involves changing the allowable value for the Turbine Condenser Vacuum – Low scram setpoint. The analytical limit, on which the transient analyses are based, is not affected. Accepted setpoint methodology was used to recalculate the allowable value while maintaining the current analytical limit. Consequently, the transient analyses are unaffected by the change.

TS Section 3.3.1.1 Required Action E.1

This change is associated with the change in RTP for which the TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions are verified not to be bypassed and is described in Section F.6 in Subsection TS SR 3.3.1.1.14 of this attachment.

F.7. TS Section 3.3.5.2, "Isolation Condenser System Instrumentation"

The proposed change involves reducing the reactor vessel high-pressure initiation time delay for the isolation condenser from ≤ 17 seconds to ≤ 15 seconds. The purpose of the isolation condenser instrumentation is to initiate actions to ensure adequate core cooling when the reactor vessel is isolated from its primary heat sink, the main condenser. At uprated power conditions, the LOFW evaluation was performed with a reduced time delay of 15 seconds as discussed in Section 3.8, "Isolation Condenser," of Attachment E. The proposed change will ensure that the isolation condenser initiates before relief valve operation reduces RPV pressure below the isoaltion condenser pressure setpoint.

F.8. TS Section 3.3.6.1, "Primary Containment Isolation Instrumentation"

TS Table 3.3.6.1-1 Function 1.d

The proposed changes increase the allowable values for the Main Steam Line Flow - High isolation functions. Since the allowable values are based on the transmitter pressure differentials and given in psid, for discussion purposes, the values will be presented in equivalent percent of rated steam flow. The allowable value for Unit 2 is increased from the equivalent of 120% of pre-uprated steam flow to 125% of post-uprated steam flow. The allowable value for Unit 3 is increased from the equivalent of 120% of pre-uprated steam flow to 140% of post-uprated steam flow. The difference in values is due to physical differences in the flow restrictors between units. Since the maximum steam flow does not change due to the flow restrictors, the proposed changes result in a decrease in the difference between the allowable value and the maximum flow. The purpose of the Main Steam Line Flow - High isolation function is to provide protection against pipe breaks in the MSL outside the drywell. For a complete severance of one MSL, steam flow increases almost instantaneously to the maximum rated steam flow as limited by the flow restrictors. Thus, the present and proposed setpoints would be attained virtually at the same time. Therefore, the consequences of a MSL break as evaluated in the UFSAR will remain unchanged with the increase in high flow setpoint. This is also discussed in Section 5.4.3, "Main Steam Line High Flow Isolation," of Attachment E.

TS Table 3.3.6.1-1 Function 2.a

This function is associated with the primary containment isolation on Reactor Vessel Water Level – Low. This change is associated with the proposed change to lower the allowable value of the RPS Reactor Vessel Water Level – Low scram function described in Section F.6 in Subsection TS Table 3.3.1.1-1 Function 4 of this attachment. The purpose of containment isolation is to minimize the potential inventory loss across the containment boundary and to prevent offsite radiation doses from exceeding 10 CFR 100 limits during a postulated LOCA. For LOCAs inside primary containment, the high drywell pressure signal will be the first signal to initiate primary containment isolation. The radiological source term is a function of the power level and the resulting fission product noble gases and iodines in the core are conservatively assumed to be immediately released following a LOCA. Thus, neither the amount of fission products released to the containment nor the time at which the containment isolates are dependent on the low water level containment isolation. For LOCAs outside containment, the main steam line break is the limiting event. This event is mitigated by the containment isolation that occurs on high steam flow or low steam line pressure. Therefore, this change does not affect the limiting event. However, small steam breaks outside containment that do not cause the isolation on high steam flow or low steamline pressure would rely on the low RPV water level isolation. Lowering of the low water level by 8 inches would not cause the mass release from the small steam break to become greater than the mass release from the large steamline break. Therefore, the delay of this isolation signal for a few seconds will not affect the ability of the containment isolation valves to perform their intended functions.

TS Table 3.3.6.1-1 Function 5.b

This function is associated with the isolation of the RWCU system on Reactor Vessel Water Level – Low. This change is associated with the proposed change to lower the allowable value of the RPS Reactor Vessel Water Level – Low scram function described in Section F.6 in Subsection TS Table 3.3.1.1-1 Function 4 of this attachment. The RWCU isolation is not directly assumed in the UFSAR safety analyses because the RWCU system line break is bounded by breaks of larger systems. This is still the case under EPU conditions. Therefore, the delay of this isolation signal for a few seconds will not affect the ability of the containment isolation valves to perform their intended functions.

TS Table 3.3.6.1-1 Function 6.b

This function is associated with the isolation of the shutdown cooling system and is only required to be operable in modes 3, 4 and 5. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some reactor vessel interfaces occurs to begin isolating the potential sources of a break. This function is not directly assumed in the safety analyses because a break in the shutdown cooling system is bounded by a break in the recirculation and main steam lines. This allowable value is being changed since it is the same as the allowable value for the RPS Reactor Vessel Water Level – Low scram function. The summary safety analysis associated with that change is described in Section F.6 in Subsection TS Table 3.3.1.1-1 Function 4 of this attachment.

F.9. TS Section 3.4.3, "Safety and Relief Valves"

The proposed change adds the safety function lift setpoint for the Target Rock safety valve to allow credit to be taken for the safety function of this valve. The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all MSIVs, followed by a reactor scram on high neutron flux. This transient assumes failure of the direct scram associated with MSIV position. This is the design basis event used to demonstrate compliance with the ASME vessel overpressure protection criteria. For the purpose of the analyses, the relief valves are not credited to function during this event and the safety mode of the Target Rock valve, which has the highest capacity, is considered out of service. The results of the analysis demonstrate that at uprated conditions, the design safety valve capacity is capable of maintaining reactor pressure below the ASME overpressure limit of 1375 for the vessel and 1345 for the dome. Since the Target Rock valve was not previously credited for use as a safety valve, the safety function lift setpoint was not included in the TS SR. Although the EPU analysis continues to assume that the Target Rock safety valve is out of service, including the safety function lift setpoint in the TS SR will allow credit to be taken for this valve in the event another safety valve is determined to be out of service. Since this valve has the highest capacity, the analysis remains bounding.

F.10. TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program"

As discussed in Section 4.1, "Containment System Performance," of Attachment E, the peak drywell pressure occurs during the short-term DBA-LOCA. The short-term DBA-LOCA analysis covers the blowdown period during which the maximum drywell pressures and differential pressures between the drywell and wetwell occur. The analysis is performed at 102% of the EPU power level, with the break flow calculated using a more detailed model that has been previously approved by the NRC. When analyzed at pre-uprate conditions using the more detailed model, the peak containment pressure is predicted to be 42.8 psig, whereas the previous model predicted a peak containment pressure of 47 psig. The EPU has a relatively insignificant impact on peak drywell pressure. The analysis predicts an increase of 1.1 psig over the pre-uprate value. The predicted peak pressure at uprated conditions of 43.9 psig is well below the maximum allowable internal pressure of 62 psig.

F.11. TS Section 5.6.5, "Core Operating Limits Report"

The proposed change deletes TS Section 5.6.5.a.4 entirely because TS Section 3.2.4 is deleted entirely. TS Section 5.6.5.a.4 requires the LHGR and the TLHGR to be included in the COLR in support of TS 3.2.4. With the removal of TS 3.2.4 as discussed in Section F.5 of this attachment, this requirement is no longer necessary. Since the TLHGR is used to calculate the FDLRC and the FDLRC calculation is removed as part of the removal of TS 3.2.4, inclusion of the TLHGR in the COLR is no longer necessary. The LHGR will continue to be included in the COLR in support of TS Section 5.6.5.a.3.

G. IMPACT ON PREVIOUS SUBMITTALS

All submittals currently under review by the NRC were evaluated to determine the impact of this submittal. The following submittals are associated with this request for amendment.

1. The TS for DNPS Unit 3 allows operation at the current RTP of 2527 MWt with the safety function of the Target Rock valve inoperable. By letter dated February 29, 2000 (Reference I.5), ComEd submitted an amendment requesting approval of an identical TS provision for DNPS Unit 2.

2. By letter dated March 3, 2000, DNPS has submitted a TS amendment request for conversion to the ITS (Reference I.6). In anticipation of approval, this request for amendment is based on the format of the ITS.

3. By letter dated September 29, 2000 (Reference I.7), DNPS submitted a request to amend the TS in response to an anticipated transition to GE14 fuel. This request for amendment included the addition of the definition of the Maximum Fraction of Limiting Power Density (MFLPD), and a proposed change to TS 3.2.4. The proposed change to TS 3.2.4 involves the inclusion of the MFLPD in the determination of the APRM Gain and Setpoint. Because of the implementation of the ARTS power and flow dependent limits, the requirements of TS 3.2.4 are deleted entirely. As a result, the definition of MFLPD is also deleted.

No other submittals currently under review by the NRC are affected by the information presented in this license amendment request.

H. SCHEDULE REQUIREMENTS

ComEd plans to fully implement the uprated power conditions for Unit 2 during the refueling outage scheduled to begin October 20, 2001 and for Unit 3 during the refueling outage scheduled to begin September 28, 2002. Therefore, ComEd requests that if found acceptable, the proposed changes be approved by October 15, 2001.

I. REFERENCES

1. GE Licensing Topical Report NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," February 1999, Proprietary, ELTR1

2. GE Licensing Topical Report NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," February 2000, Proprietary, ELTR2

3. NRC Letter, "Staff Position Concerning General Electric Boiling-Water Reactor Extended Power Uprate Program (TAC No. M91680)," February 8, 1996

4. NRC Letter, "Staff Safety Evaluation of General Electric Boiling Water Reactor (BWR) Extended Power Uprate Generic Analyses (TAC M95087)," September 14, 1998

5. Letter from Preston Swafford (ComEd) to U.S. NRC, Request for Technical Specifications Change: Reduction in the Number of Safety Valves Required for Reactor Vessel Overpressure Protection," dated February 29, 2000

6. Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power

Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated March 3, 2000

7. Letter from R.M. Krich (Commonwealth Edison Company) to U.S. NRC, "Request for Technical Specifications Change, Transition to General Electric Fuel," dated September 29, 2000

MARKED-UP TS PAGES FOR PROPOSED CHANGES

The marked-up Technical Specifications are provided in the following pages. The marked-up bases pages are also provided for reference.

REVISED LICENSE PAGES

Page 3 - Condition 2.C(1) (Unit 2)
Appendix B (Unit 2)
Page 4 – Condition 3.A (Unit 3)
Appendix B (Unit 3)

REVISED PAGES

1.1-3
1.1-4
3.2.4-1
3.2.4-2
3.3.1.1-2
3.3.1.1-4
3.3.1.1-6
3.3.1.1-8
3.3.1.1-9
3.3.1.1-10
3.3.5.2-2
3.3.6.1-5
3.3.6.1-7
3.4.3-2
5.5-12
5.6-3

Note: There are no changes on this page. This page is provided for continuity only.

NOTE: **This is a facsimile of the Dresden Nuclear Power Station License DPR-19. It will be updated whenever amendments are issued. It is currently updated through Amendment 181 dated September 27, 2000.**

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

FACILITY OPERATING LICENSE

License No. DPR-19

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

   A. The application for a license filed by the Commonwealth Edison Company (the licensee) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Dresden Nuclear Power Station, Unit 2 (the facility) has been completed in conformity with Construction Permit No. CPPR-18 and the application, as amended, the provisions of the Act, and the regulations of the Commission, and has been operating under a provisional license since December 22, 1969;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission (except as exempted from compliance in Section 2.D. below);

   D. There is reasonable assurance (i) that the activities authorized by this operating license 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 set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below);

   E. Commonwealth Edison Company is technically qualified to engage in the activities authorized by this license, as amended, in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

   F. Commonwealth Edison Company has 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 license will not be inimical to the common defense and security or to the health and safety of the public;

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

I. The receipt, possession, and use of source, byproduct and special nuclear materials as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Provisional Operating License No. DPR-19, dated December 22, 1969, as amended, is superseded in its entirety by Facility Operating License No. DPR-19 hereby issued to Commonwealth Edison Company (the licensee or CECo) to read as follows:

A. This license applies to the Dresden Nuclear Power Station, Unit 2, a boiling water reactor and associated equipment (the facility). The facility is located in Grundy County, Illinois, and is described in the licensee's Updated Final Safety Analysis Report, as supplemented and amended, and in the licensee's Environmental Report, as supplemented and amended.

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

(1) CECo, pursuant to Section 104b of the Act and 10 CFR Part 50, to possess, use and operate the facility at the designated location in Grundy County, Illinois, in accordance with the procedures and limitations set forth in this license;

(2) CECo, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

(3) CECo, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) CECo, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) CECo, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

2957

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of ~~2527~~ megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

Am. 181
09/27/00

The Technical Specifications contained in Appendix A, as revised through Amendment No. **181**, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Operation in the coastdown mode is permitted to 40% power.

(4) The valves in the equalizer piping between the recirculation loop shall be closed at all times during reactor operation.

(5) The licensee shall maintain the commitments made in response to the March 14, 1983, NUREG-0737 Order, subject to the following provision:

> The licensee may make changes to commitments made in response to the March 14, 1983, NUREG-0737 Order without prior approval of the Commission as long as the change would be permitted without NRC approval, pursuant to the requirements of 10 CFR 50.59. Consistent with this regulation, if the change results in an Unreviewed Safety Question, a license amendment shall be submitted to the NRC staff for review and approval prior to implementation of the change.

APPENDIX B

ADDITIONAL CONDITIONS

DPR

FACILITY OPERATING LICENSE NO. DRP-19

Commonwealth Edison Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
157	The license is amended to authorize changing the UFSAR to allow credit for containment overpressure as detailed below, to assure adequate Net Positive Suction Head is available for low pressure Emergency Core Cooling System pumps following a design basis accident.	Effective as of the issuance of Amendment No. 157 and shall be implemented within 30 days.

	Time (seconds)	Containment Pressure (PSIG)	
0-290	~~0-240~~	~~9.5~~	9.5
290-5000	~~240-480~~	~~2.9~~	4.8
5000-30,000	~~480-6000~~	~~1.9~~	4.25
	~~6000-accident end~~	~~2.5~~	

Amendment Number	Additional Condition	Implementation Date
157	The EOPs shall be changed to alert operator to NPSH concerns and to make containment spray operation consistent with the overpressure requirements for NPSH.	Shall be implemented within 30 days after issuance of Amendment No. 157.
160	This amendment authorizes the licensee to incorporate in the Updated Final Safety Analysis Report (UFSAR), the description of the Reactor Coolant System design pressure, temperature and volume that was removed from Technical Specification Section 5.4, and evaluated in a safety evaluation dated June 12, 1997.	30 days from the date of issuance of Amendment No. 160.
163	The licensee shall review the Dresden Operation Annunciator and General Abnormal Conditions Procedures and revise them as required to ensure operator action is taken in a timely manner to limit occupational doses and environmental releases.	60 days from the date of issuance of Amendment No. 163

NOTE: This is a facsimile of the Dresden Nuclear Power Station License DPR-25. It will be updated whenever amendments are issued. <u>It is currently updated through Amendment 176 dated September 27, 2000.</u>

<u>COMMONWEALTH EDISON COMPANY</u>

<u>DOCKET NO. 50-249</u>

<u>FACILITY OPERATING LICENSE</u>

The Atomic Energy Commission (the Commission) having found that:

a. Commonwealth Edison Company (the applicant) has submitted to the Commission all technical information required by Provisions Construction Permit No., CPPR-22, the Atomic Energy Act of 1954, as amended (the Act), and the rules and regulations of the Commission to complete the application for a construction permit and facility license dated February 10, 1966, as supplemented by application for a facility license dated November 17, 1967 and amended by Amendment Nos. 8 through 24, dated August 30, 1968, November 21, 1968, February 28, 1969, March 18, 1969, April 16, 1969, May 20, 1969, July 2, 1969, July 22, 1969, August 5, 1969, August 8, 1969, August 8, 1969, August 10, 1969, August 18, 1969, September 2, 1969, October 16, 1969, May 7, 1970, August 11, 1970 and September 4, 1970, respectively, (the application); and supplemented by the applicant's letter dated December 17, 1970, and telegram dated December 18, 1970;

b. The Dresden Nuclear Power Station Unit 3 (the facility) has been substantially completed in conformity with Provisional Construction Permit No. CPPR-22, the application, the provisions of the Act and the rules and regulations of the Commission;

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

~~DLR Authorization Dated 6-28-71~~

d. There is reasonable assurance (i) that the facility can be operated at power levels not in excess of ~~2527~~ 2957 megawatt (thermal) in accordance with this license without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

e. The applicant is technically and financially qualified to engage in the activities authorized by this operating license; in accordance with the rules and regulations of the Commission;

f. The applicant has furnished proof of financial protection to satisfy the requirements of 10 CFR Part 140;

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

Facility Operating License No. DPR-25 is hereby issued to Commonwealth Edison Company (Commonwealth Edison), as follows:

1. The license applies to the Dresden Nuclear Power Station Unit 3, a single cycle, boiling, light water reactor, and electric generating equipment (the facility). The facility is located at the Dresden Nuclear Power Station in Grundy County, Illinois, and is described in the "Safety Analysis Report," as supplemented and amended (Amendment Nos. 8 through 24).

2. Subject to the conditions and requirements incorporated herein the Commission hereby licenses Commonwealth Edison:

   A. 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 as a utilization facility at the designated location at the Dresden Nuclear Power Station;

   B. Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear materials, not including plutonium, as reactor fuel, in accordance with the limitations for storage and amounts required for operation as described in the Final Safety Analysis Report, as supplemented and amended as of September 3, 1976;

Am. 21
9/30/76

   C. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear materials as sealed neutron sources for reactor startup, scaled sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts required;

   D. Pursuant to the Act and the 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear materials without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

Am. 31 1/30/78 2. E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Dresden Nuclear Power Station, Units Nos. 1, 2 and 3.

F. Surveillance Requirements

The Surveillance Requirements contained in Appendix A Technical Specifications and listed below are not required to be performed immediately upon implementation of Amendment No. 145:

a. Surveillance Requirement 4.1.A.2 - RPS Logic System Functional Test

b. Surveillance Requirement 4.2.A.2 - Primary & Secondary Containment Logic System Functional Test

c. Surveillance Requirement 4.2.J.2 - Feedwater Pump Trip Logic System Functional Test

d. Surveillance Requirement 4.6.F.1.1.b - Relief Valve Logic System Functional Test

e. Surveillance Requirement 4.9.A.9 - Simultaneous Diesel Generator Start

f. Surveillance Requirement 4.9.A.10 - Diesel Storage Tank Cleaning (Unit 3 and Unit 2/3 only)

Each of the above Surveillance Requirements shall be successfully demonstrated prior to entering into MODE 2 on the first plant startup following the fourteenth refueling outage (D3R14).

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations; 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Level

Am. 1
3/2/71
DRL
Authorized
6/28/71

Commonwealth Edison is authorized to operate the
2957 facility at steady state power levels not in excess
of 2527 megawatts (thermal), except that Commonwealth Edison shall not operate the facility at power levels in excess of five (5) megawatts (thermal), until satisfactory completion of modifications and final testing of the station output transformer, the auto-depressurization interlock, and the feedwater system, as described in Commonwealth Edison's telegram; dated February 26, 1971, have been verified in writing by the Commission.

B. Technical Specifications

Am. 176
09/27/00

The Technical Specifications contained in Appendix A as revised through Amendment **176** are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Reports

Commonwealth Edison shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

Commonwealth Edison shall keep facility operating records in accordance with the requirements of the Technical Specifications.

Am. 94
6/20/88

E. Restrictions

Operation in the coast down mode is permitted to 40% power.

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DRP-25

Commonwealth Edison Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
152	The license is amended to authorize changing the UFSAR to allow credit for containment overpressure as detailed below, to assure adequate Net Positive Suction Head is available for low pressure Emergency Core Cooling System pumps following a design basis accident.	Prior to Unit 3 returning to Mode 3 from refueling outage D3R14.

Time (seconds)		Containment Pressure (PSIG)	
0-290	~~0-240~~	~~9.5~~	9.5
290-5000	~~240-480~~	~~2.9~~	4.8
5000-30,000	~~480-6000~~	~~1.9~~	4.25
	~~6000-accident end~~	~~2.5~~	

Amendment Number	Additional Condition	Implementation Date
152	The licensee shall complete the evaluation of the torus attached piping.	Prior to Unit 3 returning to Mode 3 from refueling outage D3R14.
152	The EOPs shall be changed to alert operator to NPSH concerns and to make containment spray operation consistent with the overpressure requirements for NPSH.	Shall be implemented within 30 days after issuance of Amendment No. 152.
155	This amendment authorizes the licensee to incorporate in the Updated Final Safety Analysis Report (UFSAR), the description of the Reactor Coolant System design pressure, temperature and volume that was removed from Technical Specification Section 5.4, and evaluated in a safety evaluation dated June 12, 1997.	30 days from the date of issuance of Amendment No. 155

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)

conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites;" Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977; or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

~~FUEL DESIGN LIMITING RATIO FOR CENTERLINE MELT (FDLRC)~~

~~The FDLRC shall be 1.2 times the LHGR existing at a given location divided by the product of the transient LHGR limit and the fraction of RTP.~~

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE; and

d. Pressure Boundary LEAKAGE

LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

(continued)

1.1 Definitions (continued)

LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components required for OPERABILITY of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE—OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~2527~~ 2957 MWt.

(continued)

Note: There are no changes on this page. This page is provided for continuity only.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Control Rod Scram Times

BASES

BACKGROUND

The scram function of the Control Rod Drive (CRD) System controls reactivity changes during anticipated operational occurrences to ensure that specified acceptable fuel design limits are not exceeded (Ref. 1). The control rods are scrammed by positive means using hydraulic pressure exerted on the CRD piston.

When a scram signal is initiated, control air is vented from the scram valves, allowing them to open by spring action. Opening the exhaust valve reduces the pressure above the main drive piston to atmospheric pressure, and opening the inlet valve applies the accumulator or reactor pressure to the bottom of the piston. Since the notches in the index tube are tapered on the lower edge, the collet fingers are forced open by cam action, allowing the index tube to move upward without restriction because of the high differential pressure across the piston. As the drive moves upward and the accumulator pressure reduces below the reactor pressure, a ball check valve opens, letting the reactor pressure complete the scram action. If the reactor pressure is low, such as during startup, the accumulator will fully insert the control rod in the required time without assistance from reactor pressure.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in Reference 2. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. The resulting negative scram reactivity forms the basis for the determination of plant thermal limits (e.g., the MCPR). Other distributions of scram times (e.g., several control rods scramming slower than the average time with several control rods scramming faster than the average time) can also provide sufficient scram reactivity. Surveillance of each individual control rod's scram time ensures the scram reactivity assumed in the DBA and transient analyses can be met.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The scram function of the CRD System protects the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and the 1% cladding plastic strain fuel design limit (see Bases for LCO ~~3.2.4, Average Power Range Monitor (APRM) Gain and Setpoint~~ 3.2.3, "Linear Heat Generation Rate"), which ensure that no fuel damage will occur if these limits are not exceeded. At ≥ 800 psig, the scram function is designed to insert negative reactivity at a rate fast enough to prevent the actual MCPR from becoming less than the MCPR SL, during the analyzed limiting power transient. Below 800 psig, the scram function is assumed to perform during the control rod drop accident (Ref. 3) and, therefore, also provides protection against violating fuel design limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control"). For the reactor vessel overpressure protection analysis, the scram function, along with the safety/relief valves, ensure that the peak vessel pressure is maintained within the applicable ASME Code limits.

Control rod scram times satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The scram times specified in Table 3.1.4-1 are required to ensure that the scram reactivity assumed in the DBA and transient analysis is met (Ref. 4). To account for single failures and "slow" scramming control rods, the scram times specified in Table 3.1.4-1 are faster than those assumed in the design basis analysis. The scram times have a margin that allows up to approximately 7% of the control rods (e.g., 177 x 7% ≈ 12) to have scram times exceeding the specified limits (i.e., "slow" control rods) assuming a single stuck control rod (as allowed by LCO 3.1.3, "Control Rod OPERABILITY") and an additional control rod failing to scram per the single failure criterion. The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when the index tube passes a specific location and then opens

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods are expected to avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 6, 7, 8, and 9. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow state ($MCPR_f$) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in UFSAR, Chapter 15 (Ref. 5).

Replace with INSERT B 3.2.2-2

~~Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 8) and a multichannel thermal hydraulic code (Ref. 9) to analyze slow flow runout transients on a cycle-specific basis. For core flows less than rated, the established MCPR operating limit is adjusted to provide protection of the MCPR SL in the event of an uncontrolled recirculation flow increase to the physical limit of the pump. Protection is provided for manual and automatic flow control by applying appropriate flow dependent MCPR operating limits. The MCPR operating limit for a given flow state is the greater of the rated conditions MCPR operating limit or the flow dependent MCPR operating limit. For automatic flow control, in addition to protecting the MCPR SL during the flow run-up event, protection is provided by the flow dependent MCPR operating limit to prevent exceeding the rated flow MCPR operating limit during an automatic flow increase to rated core flow. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.~~

The MCPR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the appropriate $MCPR_f$ or the rated condition MCPR limit.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a low recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting

(continued)

INSERT B 3.2.2-2

Flow-dependent MPCR limits, MCPR(F), ensure that the Safety Limit MCPR (SLMCPR) is not violated during recirculation flow events. The design basis flow increase event is a slow-flow power increase event which is not terminated by scram, but which stabilizes at a new core power corresponding to the maximum possible core flow. Flow runout events are simulated along a constant xenon flow control line assuming a quasi steady-state plant heat balance. The ARTS-based MCPR(F) limit is specified as an absolute value and is generic and cycle-independent. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Above the power at which the scram is bypassed, bounding power-dependent trend functions have been developed. These trend functions, K(P), are used as multipliers to the rated MCPR operating limits to obtain the power-dependent MCPR limits, MCPR(P). Below the power at which the scram is automatically bypassed, the MCPR(P) limits are actual absolute Operating Limit MCPR (OLMCPR) values. The power dependent limits are established to protect the core from plant transients other than core flow increases, including pressurization and local control rod withdrawal events.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, (i.e., steady state). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the normal operations and anticipated operating conditions identified in References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1 and 2. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 100. A mechanism that could cause fuel damage during normal operations and operational transients and that is considered in fuel evaluations is a rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient excursions above the operating limit while still remaining within the AOO limits, plus an allowance for densification power spiking.

INSERT B 3.2.3-1 →

The LHGR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

INSERT B 3.2.3-1

Flow-dependent LHGR limits, LHGRFAC(F), were designed to assure adherence to all fuel thermal-mechanical design bases in the event of slow recirculation flow runout event. From the bounding overpowers, the LHGRFAC(F) limits were derived such that during these events, the peak transient linear heat generation rate would not exceed fuel mechanical limits. The flow-dependent LHGR limits are generic, cycle-independent and are specified in terms of multipliers, LHGRFAC(F), to be applied to the rated LHGR values.

Power-dependent LHGR limits, expressed in terms of a LHGR multiplier, LHGRFAC(P), are substituted to assure adherence to the fuel thermal-mechanical design bases at reduced power conditions. The power-dependent LHGRFAC(P) limits are generated using the same database as used to determine the MCPR multiplier (K(P)). For GE fuel designs, both incipient centerline melting of the fuel and plastic strain of the cladding are considered in determining the power-dependent LHGR limit although the limiting criterion is generally incipient centerline melting. Appropriate LHGRFAC(P) limits are selected based on plant-specific transient analyses. These limits are derived to assure that peak transient LHGR for any transient is not increased above the fuel design bases.

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint

LCO 3.2.4

a. FDLRC shall be less than or equal to 1.0; or

b. Each required APRM Flow Biased Neutron Flux—High Function Allowable Value shall be modified by 1/FDLRC; or

c. Each required APRM gain shall be adjusted such that the APRM readings are ≥ 100% times the fraction of RTP (FRTP) times FDLRC.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	6 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	------------------NOTE------------------ Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4.b or LCO 3.2.4.c requirements. -- Verify FDLRC is within limits.	Once within 12 hours after ≥ 25% RTP <u>AND</u> 24 hours thereafter
SR 3.2.4.2	------------------NOTE------------------ Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4.a requirements. -- Verify each required: a. APRM Flow Biased Neutron Flux—High Function Allowable Value is modified by 1/FDLRC; or b. APRM gain is adjusted such that the APRM reading is ≥ 100% times the FRTP times FDLRC.	12 hours

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint

BASES

BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable final design criteria are discussed in UFSAR, Sections 3.1.2.2.1, 3.1.2.2.4, 3.1.2.3.1, and 3.1.2.3.10 (Ref. 1). This LCO is provided to require the APRM gain or APRM Flow Biased Neutron Flux—High Function Allowable Value (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b) to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

The condition of excessive power peaking is determined by Fuel Design Limit Ratio for Centerline Melt (FDLRC), which is defined as:

$$\text{FDLRC} = \frac{(\text{LHGR})(1.2)}{(\text{TLHGR})(\text{FRTP})} ;$$

where LHGR is the Linear Heat Generation Rate, FRTP is the Fraction of Rated Thermal Power, and TLHGR is the Transient Linear Heat Generation Rate limit. The TLHGR limit is specified in the COLR.

Maintaining FDLRC less than or equal to 1.0 ensures the fuel does not experience centerline melt during AOOs beginning at any power level and terminating at 120% RTP (APRM Fixed Neutron Flux—High Allowable Value). The APRM Flow Biased Neutron Flux—High Function Allowable Value must be adjusted to ensure that the TLHGR limit is not violated for any power distribution. When FDLRC is greater than 1.0, excessive power peaking exists. To maintain margins similar to those at RTP conditions, the APRM Flow Biased Allowable Value is decreased by 1/FDLRC. As an alternative, this adjustment may also be accomplished by increasing the gain of the APRM by FDLRC. Increasing the APRM gain raises the initial APRM reading closer to the Flow Biased Allowable Value such that a scram would be received at the same point in a transient

(continued)

BASES

BACKGROUND (continued)

as if the Allowable Value had been reduced. Thus, increasing the APRM gain by FDLRC provides the same degree of protection as reducing the APRM Flow Biased Neutron Flux—High Function Allowable Value by 1/FDLRC. Either of these adjustments has effectively the same result as maintaining FDLRC less than or equal to 1.0, and thus, maintains RTP margins for APLHGR, MCPR, and LHGR.

The normally selected APRM Flow Biased Neutron Flux—High Function Allowable Value positions the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The Allowable Value is flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow, the margin to scram is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM Allowable Value is supported by the analyses presented in Reference 3 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR, MCPR, and LHGR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the APRM Flow Biased Neutron Flux—High Function Allowable Value may be reduced during operation when FDLRC indicates an excessive power peaking distribution.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR, MCPR, and LHGR) be maintained to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

UFSAR safety analyses (Ref. 2) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR, MCPR, and LHGR) occurs. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limit the initial margins to these operating limits at rated

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of the APLHGR, the MCPR, or the LHGR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the fuel design limits could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, either the APRM Flow Biased Neutron Flux—High Function Allowable Value is adjusted downward by 1/FDLRC, or the APRM gain is adjusted upward by FDLRC. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the APRM Flow Biased Neutron Flux—High Function Allowable Value, dependent on the increased peaking that may be encountered.

The APRM gain and setpoint satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Meeting any one of the following conditions ensures acceptable operating margins for events described above:

a. Limiting excess power peaking;

b. Reducing the APRM Flow Biased Neutron Flux—High Function Allowable Value by multiplying the APRM Flow Biased Neutron Flux—High Function Allowable Value by 1/FDLRC; or

c. Increasing APRM gains to cause the APRM to read greater than or equal to 100% times FRTP times FDLRC. This condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

(continued)

BASES

LCO (continued)

Maintaining FDLRC less than or equal to 1.0 ensures the fuel does not experience centerline melt during AOOs beginning at any power level and terminating at 120% of RTP. When FDLRC is greater than 1.0, excessive power peaking exists. To compensate for this condition, the APRM Flow Biased Neutron Flux—High Function Allowable Value is adjusted downward by 1/FDLRC or the APRM gain is adjusted upward by FDLRC. When the reactor is operating with the peaking less than the design value, it is not necessary to modify the APRM Flow Biased Neutron Flux—High Function Allowable Value. Modifying the APRM Flow Biased Allowable Value or adjusting the APRM gain is equivalent to maintaining FDLRC less than or equal to 1.0, as stated in the LCO.

For compliance with LCO 3.2.4.b (APRM Flow Biased Neutron Flux—High Function Allowable Value modification) or LCO 3.2.4.c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, Function 2.b are required to be modified or adjusted. In addition, each APRM may be allowed to have its gain adjusted or Allowable Value modified independently of other APRMs that are having their gain adjusted or Allowable Value modified.

APPLICABILITY

The FDLRC limit, APRM gain adjustment, or APRM Flow Biased Neutron Flux—High Function Allowable Value is provided to ensure that the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violated during design basis transients. As discussed in the Bases for LCO 3.2.1, LCO 3.2.2, and LCO 3.2.3 sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the reactor is operating at ≥ 25% RTP.

ACTIONS

A.1

If the APRM gain or Flow Biased Neutron Flux—High Function Allowable Value is not within limits while FDLRC has exceeded 1.0, the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit may be reduced. Therefore, prompt action should be taken to restore the FDLRC to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

(continued)

BASES (continued)

ACTIONS

A.1 (continued)

The 6 hour Completion Time is normally sufficient to restore either the FDLRC to within limits or to adjust the APRM gain or modify the APRM Flow Biased Neutron Flux—High Function Allowable Value to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

B.1

If FDLRC, the APRM gain or Flow Biased Neutron Flux—High Function Allowable Value cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

The FDLRC is required to be calculated and compared to 1.0 or APRM gain adjusted or APRM Flow Biased Neutron Flux—High Function Allowable Value modified to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are only required to determine the FDLRC and, assuming FDLRC is greater than 1.0, the appropriate APRM gain or APRM Flow Biased Neutron Flux—High Function Allowable Value, and are not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or Flow Biased Neutron Flux—High Function circuitry. SR 3.2.4.1 and SR 3.2.4.2 have been modified by Notes, which clarify that the respective SR does not have to be met if the alternate requirement demonstrated by the other SR is satisfied. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1), MCPR (LCO 3.2.2), and LHGR (LCO 3.2.3). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2 (continued)

12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to APLHGR, MCPR, and LHGR operating limits at low power levels.

The 12 hour Frequency of SR 3.2.4.2 is required when FDLRC is greater than 1.0, because more rapid changes in power distribution are typically expected.

REFERENCES

1. UFSAR, Sections 3.1.2.2.1, 3.1.2.2.4, 3.1.2.3.1, and 3.1.2.3.10.
2. UFSAR, Chapter 15.

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

-----NOTES-----

1. Separate Condition entry is allowed for each channel.

2. When Function 2.b and 2.c channels are inoperable due to APRM indication not within limits, entry into associated Conditions and Required Actions may be delayed for up to 2 hours if the APRM is indicating a lower power value than the calculated power, and for up to 12 hours if the APRM is indicating a higher power value than the calculated power.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours
	<u>OR</u>	
	A.2 Place associated trip system in trip.	12 hours
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip.	6 hours
	<u>OR</u>	
	B.2 Place one trip system in trip.	6 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1 Reduce THERMAL POWER to < ~~45%~~ 38.5% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Be in MODE 2. <u>AND</u> F.2 ---------NOTE-------- Only required to be met for Function 5, Main Steam Isolation Valve—Closure, and Function 10, Turbine Condenser Vacuum—Low. --------------------- Reduce reactor pressure to < 600 psig.	8 hours 8 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1 Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTES-----------------------------------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	------------------NOTE------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. -- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP. ~~plus any gain adjustment required by LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint" while operating at ≥ 25% RTP.~~	7 days
SR 3.3.1.1.3	Adjust the channel to conform to a calibrated flow signal.	7 days

(continued)

Note: There are no changes on this page. This page is provided for continuity only.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.4	-------------------NOTE------------------- Not required to be performed when entering MODE 2 from MODE 1 until 24 hours after entering MODE 2. --- Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.5	Perform a functional test of each RPS automatic scram contactor.	7 days
SR 3.3.1.1.6	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to fully withdrawing SRMs
SR 3.3.1.1.7	-------------------NOTE------------------- Only required to be met during entry into MODE 2 from MODE 1. --- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.1.1.9	Calibrate the local power range monitors.	2000 effective full power hours
SR 3.3.1.1.10	Perform CHANNEL CALIBRATION.	31 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.11	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.12	Calibrate the trip units.	92 days
SR 3.3.1.1.13	Perform CHANNEL CALIBRATION.	92 days
SR 3.3.1.1.14	Verify Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ ~~45%~~ 38.5% RTP.	92 days
SR 3.3.1.1.15	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 24 hours after entering MODE 2. 3. For Function 2.b, not required for the flow portion of the channels. --- Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.16	Perform CHANNEL FUNCTIONAL TEST.	24 months

(continued)

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.17	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 24 hours after entering MODE 2. --- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.18	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.19	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 5 "n" equals 4 channels for the purpose of determining the the STAGGERED TEST BASIS Frequency. --- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux – High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 121/125 divisions of full scale
	5[(a)]	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 121/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.18	NA
	5[(a)]	3	H	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.18	NA
2. Average Power Range Monitors					
a. Neutron Flux – High, Setdown	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.18	≤ 17.1% RTP
b. Flow Biased Neutron Flux – High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	0.56 W + 67.4% ≤ ~~0.58 W + 63.5%~~ RTP and ≤ 122% RTP[(b)]

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) ~~0.58 W + 59.2% and ≤ 118.5%~~ RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." 0.56 W + 63.2% and ≤ 118.5%

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Fixed Neutron Flux – High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 122% RTP
d. Inop	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.18	NA
3. Reactor Vessel Steam Dome Pressure – High	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 1054 psig
4. Reactor Vessel Water Level – Low	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≥ ~~10.24~~ inches 2.65
5. Main Steam Isolation Valve – Closure	1, 2(c)	8	F	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 9.5% closed
6. Drywell Pressure – High	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 1.94 psig

(continued)

(c) With reactor pressure ≥ 600 psig.

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Volume Water Level – High					
a. Thermal Switch (Unit 2) Float Switch (Unit 3)	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 37.9 gallons (Unit 2) ≤ 39.1 gallons (Unit 3)
	5[(a)]	2	H	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 37.9 gallons (Unit 2) ≤ 39.1 gallons (Unit 3)
b. Differential Pressure Switch	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 37.9 gallons (Unit 2) ≤ 39.1 gallons (Unit 3)
	5[(a)]	2	H	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 37.9 gallons (Unit 2) ≤ 39.1 gallons (Unit 3)
8. Turbine Stop Valve – Closure	38.5% ≥ ~~45%~~ RTP	4	E	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.14 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 9.5% closed
9. Turbine Control Valve Fast Closure, Trip Oil Pressure – Low	38.5% ≥ ~~45%~~ RTP	2	E	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.14 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≥ 466 psig
10. Turbine Condenser Vacuum – Low	1, 2[(c)]	2	F	SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.18 SR 3.3.1.1.19	21.4 ≥ ~~21.15~~ inches Hg vacuum
11. Reactor Mode Switch – Shutdown Position	1,2	1	G	SR 3.3.1.1.16 SR 3.3.1.1.18	NA
	5[(a)]	1	H	SR 3.3.1.1.16 SR 3.3.1.1.18	NA
12. Manual Scram	1,2	1	G	SR 3.3.1.1.8 SR 3.3.1.1.18	NA
	5[(a)]	1	H	SR 3.3.1.1.8 SR 3.3.1.1.18	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(c) With reactor pressure ≥ 600 psig.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. Reactor Vessel Steam Dome Pressure—High

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. ~~No specific safety analysis takes direct credit for this Function. However, the~~ The Reactor Vessel Steam Dome Pressure—High Function initiates a scram for transients that results in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analysis of Reference 2, reactor scram (the analyses conservatively assume scram on the Average Power Range Monitor Fixed Neutron Flux—High signal, not the Reactor Vessel Steam Dome Pressure—High or the Main Steam Isolation Valve—Closure signals), along with the safety valves, limits the peak RPV pressure to less than the ASME Section III Code limits.

High reactor pressure signals are initiated from four pressure switches that sense reactor pressure. The Reactor Vessel Steam Dome Pressure—High Allowable Value is chosen to provide a sufficient margin to the ASME Section III Code limits during the event.

Four channels of Reactor Vessel Steam Dome Pressure—High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required to be OPERABLE in MODES 1 and 2 when the RCS is pressurized and the potential for pressure increase exists.

4. Reactor Vessel Water Level—Low

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, a reactor scram is initiated at this level to substantially reduce the heat generated in the fuel from fission. The Reactor Vessel Water Level—Low Function is assumed in the analysis of the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

8. Turbine Stop Valve—Closure

Closure of the TSVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated at the start of TSV closure in anticipation of the transients that would result from the closure of these valves. The Turbine Stop Valve—Closure Function is the primary scram signal for the turbine trip event analyzed in Reference 11. For this event, the reactor scram reduces the amount of energy required to be absorbed and ensures that the MCPR SL is not exceeded.

Turbine Stop Valve—Closure signals are initiated from position switches located on each of the four TSVs. A position switch and two independent contacts are associated with each stop valve. One of the two contacts provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve—Closure channels, each consisting of one position switch (which is common to a channel in the other RPS trip system) and a switch contact. The logic for the Turbine Stop Valve—Closure Function is such that three or more TSVs must be closed to produce a scram. This Function must be enabled at THERMAL POWER ≥ ~~45%~~ 38.5% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect the OPERABILITY of this Function.

The Turbine Stop Valve—Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Stop Valve—Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function even if one TSV should fail to close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ ~~45%~~ 38.5% RTP. This Function is not required when THERMAL POWER is < ~~45%~~ 38.5% RTP since the Reactor Vessel Steam Dome Pressure—High and the Average Power Range Monitor Fixed Neutron Flux—High Functions are adequate to maintain the necessary safety margins.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

9. Turbine Control Valve Fast Closure, Trip Oil Pressure – Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure – Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 12. For this event, the reactor scram reduces the amount of energy required to be absorbed and ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, Trip Oil Pressure – Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure switch is associated with each control valve, and the signal from each switch is assigned to a separate RPS logic channel. 38.5%
This Function must be enabled at THERMAL POWER ≥ 45% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect the OPERABILITY of this Function.

The Turbine Control Valve Fast Closure, Trip Oil Pressure – Low Allowable Value is selected high enough to detect imminent TCV fast closure.

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure – Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis 38.5%
assumptions, whenever THERMAL POWER is ≥ 45% RTP. This Function is not required when THERMAL POWER is < 45% RTP, 38.5%
since the Reactor Vessel Steam Dome Pressure – High and the Average Power Range Monitor Fixed Neutron Flux – High Functions are adequate to maintain the necessary safety margins.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the RPS will trip when necessary.

SR 3.3.1.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. ~~LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint," allows the APRMs to be reading greater than actual THERMAL POWER to compensate for localized power peaking. When this adjustment is made, the requirement for the APRMs to indicate within 2% RTP of calculated power is modified to~~

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.2 (continued)

~~require the APRMs to indicate within 2% RTP of the calculated value established by SR 3.2.4.2.~~ The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.9.

An allowance is provided that requires the SR to be performed only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR, APLHGR, and LHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

SR 3.3.1.1.3

The Average Power Range Monitor Flow Biased Neutron Flux—High Function uses the recirculation loop drive flows to vary the trip setpoint. This SR ensures that the total loop drive flow signals from the flow converters used to vary the setpoint is appropriately compared to a calibrated flow signal and, therefore, the APRM Function accurately reflects the required setpoint as a function of flow. Each flow signal from the respective flow converter must be ≤ 100% of the calibrated flow signal. If the flow converter signal is not within the limit, all required APRMs that receive an input from the inoperable flow converter must be declared inoperable.

The Frequency of 7 days is based on engineering judgment, operating experience, and the reliability of this instrumentation.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.11 and SR 3.3.1.1.16 (continued)

Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The 92 day Frequency of SR 3.3.1.1.11 is based on the reliability analysis of Reference 13. The 24 month Frequency of SR 3.3.1.1.16 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.1.1.12

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.1.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference 13.

SR 3.3.1.1.14

This SR ensures that scrams initiated from the Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ ~~45%~~ RTP. This involves calibration of the bypass channels. Adequate margins for

33.5%

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.14 (continued)

the instrument setpoint methodologies are incorporated into the Allowable Value and the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must remain closed during an in-service calibration at THERMAL POWER ≥ ~~45%~~ 38.5% RTP, if performing the calibration using actual turbine first stage pressure, to ensure that the calibration remains valid.

If any bypass channels setpoint is nonconservative (i.e., the Functions are bypassed at ≥ ~~45%~~ 38.5% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel is considered OPERABLE.

The Frequency of 92 days is based on engineering judgment and reliability of the components.

SR 3.3.1.1.18

The LOGIC SYSTEM FUNCTIONAL TEST (LSFT) demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods (LCO 3.1.3, "Control Rod Operability"), and SDV vent and drain valves (LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves"), overlaps this Surveillance to provide complete testing of the assumed safety function.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.2.2 Feedwater System and Main Turbine High Water Level Trip Instrumentation

BASES

BACKGROUND

The Feedwater System and Main Turbine High Water Level Trip Instrumentation is designed to detect a potential failure of the Feedwater Level Control System that causes excessive feedwater flow.

With excessive feedwater flow, the water level in the reactor vessel rises toward the high water level reference point, causing the trip of the three feedwater pumps and the main turbine.

Reactor Vessel Water Level—High signals are provided by level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level in the reactor vessel (variable leg). Four channels of Reactor Vessel Water Level—High instrumentation are provided as input to two trip systems. Each trip system is arranged with a two-out-of-two initiation logic that trips the three feedwater pumps and the main turbine. The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a feedwater pump and main turbine trip signal to the trip logic.

A trip of the feedwater pumps limits further increase in reactor vessel water level by limiting further addition of feedwater to the reactor vessel. A trip of the main turbine and closure of the stop valves protects the turbine from damage due to water entering the turbine.

APPLICABLE SAFETY ANALYSES

The Feedwater System and Main Turbine High Water Level Trip Instrumentation is assumed to be capable of providing a feedwater pump and main turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event (Ref. 1). The high level trip

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

38.5%
indirectly initiates a reactor scram from the main turbine trip (above ~~45%~~ RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR.

Feedwater System and Main Turbine High Water Level Trip Instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires four channels of the Reactor Vessel Water Level – High instrumentation to be OPERABLE to ensure that no single instrument failure will prevent the feedwater pumps and main turbine trip on a valid high level signal. Two channels are needed to provide trip signals in order for the feedwater pump and main turbine trips to occur. Each channel must have its setpoint set within the specified Allowable Value of SR 3.3.2.2.4. The Allowable Value is set to ensure that the thermal limits are not exceeded during the event. The actual setpoint is calibrated to be consistent with the applicable setpoint methodology assumptions. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The trip setpoints are determined from the analytic limits, corrected for defined process, calibration, and instrument errors. The Allowable Values are then determined, based on the trip setpoint values, by accounting for the calibration based errors. These calibration based errors are limited to reference accuracy, instrument drift, errors associated with

(continued)

BASES

LCO (continued)

measurement and test equipment, and calibration tolerance of loop components. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrument uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for and appropriately applied for the instrumentation.

APPLICABILITY

The Feedwater System and Main Turbine High Water Level Trip Instrumentation is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE," and LCO ~~3.2.4~~, "Average ~~Power~~ Range ~~Monitor (APRM)~~ ~~Gain and~~ Setpoint," sufficient margin to these limits exists below 25% RTP; therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A Note has been provided to modify the ACTIONS related to Feedwater System and Main Turbine High Water Level Trip Instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable Feedwater System and Main Turbine High Water Level Trip Instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable Feedwater System and Main Turbine High Water Level Trip Instrumentation channel.

(continued)

3.3 INSTRUMENTATION

3.3.5.2 Isolation Condenser (IC) System Instrumentation

LCO 3.3.5.2 Four channels of Reactor Vessel Pressure—High instrumentation shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > 150 psig.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Reactor Vessel Pressure—High channels inoperable.	A.1 Declare IC System inoperable. <u>AND</u> A.2 Place channel(s) in trip.	1 hour from discovery of loss of IC initiation capability 24 hours
B. Required Action and associated Completion Time not met.	B.1 Declare IC System inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the Reactor Vessel Pressure—High Function maintains IC initiation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.5.2.1	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.5.2.2	-----------------NOTE------------------ Not required for the time delay portion of the channel. --------------------------------------- Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ 1064 psig.	 92 days
SR 3.3.5.2.3	Perform CHANNEL CALIBRATION for the time delay portion of the channel. The Allowable Value shall be ≤ ~~17~~ 15 seconds.	24 months
SR 3.3.5.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

B 3.3 INSTRUMENTATION

B 3.3.5.2 Isolation Condenser (IC) System Instrumentation

BASES

BACKGROUND

The purpose of the IC System instrumentation is to initiate actions to ensure adequate core cooling when the reactor vessel is isolated from its primary heat sink (the main condenser). A more complete discussion of IC System operation is provided in the Bases of LCO 3.5.3, "IC System."

The IC System may be initiated by either automatic or manual means. Automatic initiation occurs for sustained (~~about 17~~ ~~seconds~~) conditions of reactor vessel pressure high. The variable is monitored by four pressure switches that are connected to four time delay relays. The outputs of the time delay relays are connected in a one-out-of-two logic to a trip relay. The output of the trip relays are connected in a two-out-of-two logic arrangement. Once initiated, the IC logic can be overridden by the operator.

APPLICABLE SAFETY ANALYSES

The function of the IC System to provide core cooling to the reactor is used to respond to a main steam line isolation event. The IC System is not an Engineered Safety Feature System. and no ~~credit~~ ~~is taken~~ in ~~the safety analyses~~ for ~~IC~~ System ~~operation~~. Based on its contribution to the reduction of overall plant risk, however, the IC System, and therefore its instrumentation, satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The OPERABILITY of the IC System instrumentation is dependent upon the OPERABILITY of the four channels of the Reactor Vessel Pressure—High Function. Each channel must have its setpoint within the Allowable Value specified in SR 3.3.5.2.2. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

The Allowable Value for the IC System instrumentation Function is specified in the SR. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the

(continued)

3.3 INSTRUMENTATION

3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours for Functions 1.a, 2.a, 2.b, 5.b, and 6.b <u>AND</u> 24 hours for Functions other than Functions 1.a, 2.a, 2.b, 5.b, and 6.b
B. One or more automatic Functions with isolation capability not maintained.	B.1 Restore isolation capability.	1 hour

(continued)

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level – Low Low	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ -56.77 inches
b. Main Steam Line Pressure – Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≥ 831 psig
c. Main Steam Line Pressure – Timer	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 0.280 seconds (Unit 2) ≤ 0.236 seconds (Unit 3)
d. Main Steam Line Flow – High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ ~~160.5~~ 259.2 psid (Unit 2) ≤ ~~117.1~~ 252.6 psid (Unit 3)
e. Main Steam Line Tunnel Temperature – High	1,2,3	2 per trip string	D	SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 200°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level – Low	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ ~~10.24~~ 2.65 inches
b. Drywell Pressure – High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 1.81 psig
c. Drywell Radiation – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 77 R/hr

(continued)

Table 3.3.6.1-1 (page 2 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 290.16% of rated steam flow (Unit 2) ≤ 288.23% of rated steam flow (Unit 3)
b. HPCI Steam Line Flow – Timer	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 3.2 seconds and ≤ 8.8 seconds
c. HPCI Steam Supply Line Pressure – Low	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 104 psig
d. HPCI Turbine Area Temperature – High	1,2,3	4(a)	F	SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 189°F
4. Isolation Condenser System Isolation					
a. Steam Flow – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 290.76% of rated steam flow
b. Return Flow – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 30.2 inches water (Unit 2) ≤ 13.7 inches water (Unit 3)

(continued)

(a) All four channels must be associated with a single trip string.

Table 3.3.6.1-1 (page 3 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup System Isolation					
a. SLC System Initiation	1,2	1	H	SR 3.3.6.1.7	NA
b. Reactor Vessel Water Level – Low	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ ~~10.24~~ inches 2.65
6. Shutdown Cooling System Isolation					
a. Recirculation Line Water Temperature – High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 346°F
b. Reactor Vessel Water Level – Low	3,4,5	2(b)	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ ~~10.24~~ inches 2.65

(b) In MODES 4 and 5, provided Shutdown Cooling System integrity is maintained, only one channel per trip system with an isolation signal available to one shutdown cooling pump suction isolation valve is required.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety and Relief Valves

LCO 3.4.3 The safety function of 8 safety valves shall be OPERABLE.

AND

The relief function of 5 relief valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One relief valve inoperable.	A.1 Restore the relief valve to OPERABLE status.	14 days
B. Required Action and associated Completion Time of Condition A not met. OR Two or more relief valves inoperable. OR One or more safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the safety valves are as follows: Number of Safety Valves / Setpoint (psig) 1 / 1135 ± 11.3 2 / 1240 ± 12.4 2 / 1250 ± 12.5 4 / 1260 ± 12.6	In accordance with the Inservice Testing Program
SR 3.4.3.2	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify each relief valve opens when manually actuated.	24 months
SR 3.4.3.3	-------------------NOTE------------------- Valve actuation may be excluded. -- Verify each relief valve actuates on an actual or simulated automatic initiation signal.	24 months

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 Safety and Relief Valves

BASES

BACKGROUND

The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of safety valves are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB). Each unit is designed with nine safety valves, one of which also functions in the relief mode. This valve is a dual function Target Rock safety/relief valve (S/RV).

The safety valves and S/RV are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The safety valves actuate in the safety mode (or spring mode of operation). In this mode, the safety valve opens when the inlet steam pressure reaches the lift set pressure. At that point, the vertical upward force generated by the inlet pressure under the valve disc balances the downward force generated by the spring. Slight steam leakage develops across the valve disc-to-seat interface and is directed into the huddle chamber. Pressure builds up rapidly in the huddle chamber developing an additional vertical lifting force on the disc and disc holder. This additional force in conjunction with the expansive characteristic of steam causes the valve to "pop" open to almost full lift. This satisfies the Code requirement. The S/RV is a dual function Target Rock valve that can actuate by either of two modes: the safety mode or the relief mode. In the safety mode (or spring mode of operation), the S/RV spring loaded pilot valve opens when steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. Opening the pilot valve allows a pressure differential to develop across the main valve piston and opens the main valve. In the relief mode (or power actuated mode of operation), automatic or manual switch actuation energizes a solenoid valve which pneumatically actuates a plunger located within the main valve body. Actuation of the plunger allows pressure to be vented from the top of the main valve piston. This allows reactor pressure to lift the main valve piston, which opens the main valve. The relief valves and S/RV discharge steam

(continued)

BASES

BACKGROUND (continued)

through a discharge line to a point below the minimum water level in the suppression pool. The safety valves discharge directly to the drywell.

In addition to the safety valves and S/RV, each unit is designed with four relief valves which actuate in the relief mode to control RCS pressure during transient conditions to prevent the need for safety valve actuation (except S/RV) following such transients. The relief valves are also located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. ~~These valves are sized by assuming a turbine trip, a coincident scram and a failure of the turbine bypass system.~~ The relief valves are of the Electromatic type, which are opened by automatic or manual switch actuation of a solenoid. The switch energizes the solenoid to actuate a plunger, which contacts the pilot valve operating lever, thereby opening the pilot valve. When the pilot valve opens, pressure under the main valve disc is vented. This allows reactor pressure to overcome main valve spring pressure, which forces the main valve disc downward to open the main valve. Two of the five relief valves are the low set relief valves and all of the relief valves, including the S/RV, are Automatic Depressurization System (ADS) valves. The low set relief requirements are specified in LCO 3.6.1.6, "Low Set Relief Valves," and the ADS requirements are specified in LCO 3.5.1, "ECCS—Operating."

APPLICABLE SAFETY ANALYSES

The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs), followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. 1). For the purpose of the analyses, eight safety valves are assumed to operate in the safety mode. The relief valves and S/RV are not credited to function during this event. The analysis results demonstrate that the design safety valve capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig is met during the Design Basis Event.

(continued)

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BASES

APPLICABLE SAFETY ANALYSES (continued)

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. For other pressurization events, such as a turbine trip or generator load rejection with Main Turbine Bypass System failure (Refs. 2 and 3, respectively), the relief valves as well as the S/RV are assumed to function. The opening of the relief valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MINIMUM CRITICAL POWER RATIO (MCPR) during these events. In these events, the operation of four of the five relief valves are required to mitigate the events. Reference 4 discusses additional events that are expected to actuate the safety and relief valves.

Safety and relief valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The safety function of eight safety valves are required to be OPERABLE to satisfy the assumptions of the safety analysis (Ref. 1). The safety valve requirements of this LCO are applicable to the capability of the safety valves to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety function).

The safety valve setpoints are established to ensure that the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve setpoint to be at or below vessel design pressure (1250 psig) and the highest safety valve to be set so that the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. The transient evaluations in the UFSAR are based on these setpoints, but also include the additional uncertainties of ± 1% of the nominal setpoint drift to provide an added degree of conservatism.

Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.

(continued)

BASES

LCO (continued)

The relief valves, including the S/RV, are required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that MCPR is not exceeded.

APPLICABILITY

In MODES 1, 2, and 3, eight safety valves ~~(not including the S/RV)~~ and five relief valves ~~(including the S/RV)~~ must be OPERABLE, since considerable energy may be in the reactor core and the limiting design basis transients are assumed to occur in these MODES. The safety and relief valves may be required to provide pressure relief to discharge energy from the core until such time that the Shutdown Cooling (SDC) System is capable of dissipating the core heat.

In MODE 4, decay heat is low enough for the Shutdown Cooling System to provide adequate cooling, and reactor pressure is low enough that the overpressure and MCPR limits are unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The safety and relief functions are not needed during these conditions.

ACTIONS

<u>A.1</u>

the relief function of the

With the relief function of one relief valve (or S/RV) inoperable, the remaining OPERABLE relief valves are capable of providing the necessary protection. However, the overall reliability of the pressure relief system is reduced because additional failures in the remaining OPERABLE relief valves could result in failure to adequately relieve pressure during a limiting event. For this reason, continued operation is permitted for a limited time only.

The 14 day Completion Time to restore the inoperable required relief valve to OPERABLE status is based on the relief capability of the remaining relief valves, the low probability of an event requiring relief valve actuation, and a reasonable time to complete the Required Action.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Reactor Steam Dome Pressure

BASES

BACKGROUND	The reactor steam dome pressure is an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria and is also an assumed initial condition of design basis accidents and transients.
APPLICABLE SAFETY ANALYSES	The reactor steam dome pressure of ≤ 1005 psig is an initial condition of the vessel overpressure protection analysis of Reference 1. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analyses are conserved. Reference 2 also assumes an initial reactor steam dome pressure for the analyses of design basis accidents and transients used to determine the limits for fuel cladding integrity (see Bases for LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and 1% cladding plastic strain (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)~~,~~" ~~and LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint"~~). The nominal reactor operating pressure is approximately 1005 psig. Transient analyses typically use the nominal or a design dome pressure as input to the analysis. Small deviations (5 to 10 psi) from the nominal pressure are not expected to change most of the transient analyses results. However, sensitivity studies for fast pressurization events (main turbine generator load rejection without bypass, turbine trip without bypass, and feedwater controller failure) indicate that the delta-CPR may increase for lower initial pressures. Therefore, the fast pressurization events have considered a bounding initial pressure based on a typical operating range to assure a conservative delta-CPR and operating limit. Reactor steam dome pressure satisfies the requirements of Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

Note: There are no changes on this page. This page is provided for continuity only.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND ISOLATION CONDENSER (IC) SYSTEM

B 3.5.3 IC System

BASES

BACKGROUND

The IC System is not part of the ECCS; however, the IC System is included with the ECCS section because of their similar functions.

The IC System is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation to provide adequate core cooling. Under these conditions, the High Pressure Coolant Injection (HPCI) and IC systems perform similar functions.

The IC System (Ref.1) is a passive high pressure system comprised of one natural circulation heat exchanger, two AC motor-operated isolation valves, two D.C. motor-operated isolation valves, and two tube side high point vent isolation valves to main steam line "A". The IC System functions as a heat sink for decay heat removal from the reactor vessel following reactor scram and isolation from the main condenser. This function prevents overheating of the reactor fuel, controls reactor pressure, and limits the loss of reactor coolant through the relief valves. The IC System is automatically initiated by sustained reactor vessel high pressure and, once activated, remains in operation until manually removed from service.

The isolation condenser shell contains two tube bundles. When the IC System is in operation, both tube bundles are in service.

The IC System is designed to provide core cooling for reactor pressure ≥ 150 psig. The shell side of the condenser has a minimum water level of 6 feet which provides an inventory of ≥ 18,700 gallons. This minimum level provides ≥ 11,300 gallons (approximately 3 feet) of water above the top of the tube bundles. The shell side water temperature must be ≤ 210°F. During normal plant operations, when the system is in standby, makeup is from the clean demineralized water storage tank. Makeup during IC System operation can be provided from the Condensate

(continued)

Note: There are no changes on this page. This page is provided for continuity only.

BASES

BACKGROUND (continued)	Transfer System. Since during operation of the IC System, water in the shell will boil, the condenser is vented to the atmosphere via one line.
APPLICABLE SAFETY ANALYSES	The function of the IC System is to respond to main steam line isolation events by providing core cooling to the reactor. Although the IC System is an Engineered Safety Feature System, no credit is taken in the accident analyses for IC System operation. Based on its contribution to the reduction of overall plant risk, the system satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).
LCO	The OPERABILITY of the IC System provides adequate core cooling such that actuation of any of the low pressure ECCS subsystems is not required in the event of RPV isolation. The IC System reduces the loss of RPV inventory during an isolation event.
APPLICABILITY	The IC System is required to be OPERABLE during MODE 1, and MODES 2 and 3 with reactor steam dome pressure > 150 psig, since IC is the primary non-ECCS source for core cooling when the reactor is isolated and pressurized. In MODES 2 and 3 with reactor steam dome pressure ≤ 150 psig, and in MODES 4 and 5, IC is not required to be OPERABLE since the low pressure ECCS injection/spray subsystems can provide sufficient core cooling.
ACTIONS	<u>A.1 and A.2</u> If the IC System is inoperable during MODE 1, or MODE 2 or 3 with reactor steam dome pressure > 150 psig, and the HPCI System is immediately verified to be OPERABLE, the IC System must be restored to OPERABLE status within 14 days. In this Condition, loss of the IC System will not affect the overall plant capability to provide makeup inventory at high reactor pressure since the HPCI System is the only high pressure system assumed to function during a loss of coolant accident (LOCA). OPERABILITY of HPCI is therefore verified immediately when the IC System is inoperable. This may be performed as an administrative check, by examining logs or

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

other information, to determine if HPCI is out of service for maintenance or other reasons. It does not mean it is necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the HPCI System. If the OPERABILITY of the HPCI System cannot be immediately verified, however, Condition B must be immediately entered. For transients and certain abnormal events with no LOCA, IC (as opposed to HPCI) is an acceptable source of core cooling which also limits the loss of the RPV water level. Therefore, a limited time is allowed to restore the inoperable IC to OPERABLE status.

The 14 day Completion Time is based on a reliability study (Ref. 2) that evaluated the impact on ECCS availability, assuming various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowed outage times (AOTs). Because of similar functions of HPCI and IC, the AOTs (i.e., Completion Times) determined for HPCI are also applied to IC.

B.1 and B.2

If the IC System cannot be restored to OPERABLE status within the associated Completion Time, or if the HPCI System is simultaneously inoperable, the plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and reactor steam dome pressure reduced to ≤ 150 psig within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

This SR verifies the water volume and temperature in the shell side of the IC to be sufficient for proper operation. Based on a scram from ~~2552.3~~ 3016 MWt (~~101%~~ 102% RTP), a minimum water level of 6 feet at a temperature of ≤ 210°F in the condenser

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1 (continued)

provides sufficient decay heat removal capability for 20 minutes of operation without makeup water, before beginning to uncover the tube bundles. The volume and temperature allow sufficient time for the operator to provide makeup to the condenser.

The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during normal operation.

SR 3.5.3.2

Verifying the correct alignment for manual, power operated, and automatic valves in the IC flow path provides assurance that the proper flow path will exist for IC operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency of this SR was derived from the Inservice Testing Program requirements for performing valve testing at least once every 92 days. The Frequency of 31 days is further justified because the valves are operated under procedural control and because improper valve position would affect only the IC System. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.3.3

The IC System is required to actuate automatically in order to verify its design function satisfactorily. This Surveillance verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of the IC System will cause the system to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.3 (continued)

operate as designed; that is, actuation of all automatic valves to their required positions. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.2 overlaps this Surveillance to provide complete testing of the assumed design function.

The 24 month Frequency is based on the need to perform the Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.5.3.4

Verifying the proper flow path and heat exchange capacity for IC System operation ensures the capability of the IC System to remove the design heat load. This SR verifies the IC System capability to remove heat consistent with the design requirements of 252.5 x 10^6 Btu/hr. The IC System capacity is equivalent to the decay heat rate ~~5 minutes~~ about 530 seconds (8.8 minutes) after a reactor scram.

The 60 month Frequency is based on engineering judgement, and has been shown to be acceptable through operating experience.

REFERENCES

1. UFSAR, Section 5.4.6.

2. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Pressure

BASES

BACKGROUND

The drywell pressure is limited during normal operations to preserve the initial conditions assumed in the accident analysis for a Design Basis Accident (DBA) or loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

Primary containment performance is evaluated for the entire spectrum of break sizes for postulated LOCAs (Ref. 1). Among the inputs to the DBA is the initial primary containment internal pressure (Ref. 1). Analyses assume an initial drywell pressure of 1.5 psig. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell internal pressure does not exceed the maximum allowable of 62 psig.

The maximum calculated drywell pressure occurs during the reactor blowdown phase of the DBA, which assumes an instantaneous recirculation line break. The calculated peak drywell pressure for this limiting event is 48 psig
43.9
(Ref. 1).

Drywell pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

In the event of a DBA, with an initial drywell pressure ≤ 1.5 psig, the resultant peak drywell accident pressure will be maintained below the drywell design pressure.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining drywell pressure within limits is not required in MODE 4 or 5.

(continued)

BASES (continued)

ACTIONS

A.1

With drywell pressure not within the limit of the LCO, drywell pressure must be restored within 1 hour. The Required Action is necessary to return operation to within the bounds of the primary containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, "Primary Containment," which requires that primary containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If drywell pressure cannot be restored to within the limit within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.4.1

Verifying that drywell pressure is within the limit ensures that unit operation remains within the limit assumed in the primary containment analysis. The 12 hour Frequency of this SR was developed, based on operating experience related to trending of drywell pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal drywell pressure condition.

REFERENCES

UFSAR Section 6.2.1.3

1. Dresden ~~Nuclear~~ Power ~~Station Units~~ 2 and 3 ~~Plant~~ ~~Unique~~ Analysis ~~Report~~, COM-02-041, ~~May~~ 1983.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.1 Suppression Pool Average Temperature

BASES

BACKGROUND

The suppression chamber is a toroidal shaped, steel pressure vessel containing a volume of water called the suppression pool. The suppression pool is designed to absorb the decay heat and sensible energy released during a reactor blowdown from relief valve discharges or from Design Basis Accidents (DBAs). The suppression pool must quench all the steam released through the downcomer lines during a loss of coolant accident (LOCA). This is the essential mitigative feature of a pressure suppression containment that ensures that the peak containment pressure is maintained below the maximum allowable pressure for DBAs (62 psig). The suppression pool must also condense steam from steam exhaust lines in the turbine driven High Pressure Coolant Injection System. Suppression pool average temperature (along with LCO 3.6.2.2, "Suppression Pool Water Level") is a key indication of the capacity of the suppression pool to fulfill these requirements.

The technical concerns that lead to the development of suppression pool average temperature limits are as follows:

a. Complete steam condensation;

b. Primary containment peak pressure and temperature;

c. Condensation oscillation loads; and

d. Chugging loads.

APPLICABLE SAFETY ANALYSES

The postulated DBA against which the primary containment performance is evaluated is the entire spectrum of postulated pipe breaks within the primary containment. Inputs to the safety analyses (Reference 1) include initial suppression pool water volume and suppression pool temperature. ~~(Reference 1 for LOCAs and Reference 2 for the pool temperature analyses required by Reference 3).~~ An initial pool temperature of 95°F is assumed for the Reference 1~~, 2, and 4~~ analyses. Reactor shutdown at a pool temperature of

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

Continued addition of heat to the suppression pool with suppression pool temperature > 120°F could result in exceeding the design basis maximum allowable values for primary containment temperature or pressure. Furthermore, if a blowdown were to occur when the temperature was > 120°F, the maximum allowable bulk and local temperatures could be exceeded very quickly.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1.1

The suppression pool average temperature is regularly monitored to ensure that the required limits are satisfied. The average temperature is determined by taking an arithmetic average of OPERABLE suppression pool water temperature channels. The 24 hour Frequency has been shown, based on operating experience, to be acceptable. When heat is being added to the suppression pool by testing, however, it is necessary to monitor suppression pool temperature more frequently. The 5 minute Frequency during testing is justified by the rates at which tests will heat up the suppression pool, has been shown to be acceptable based on operating experience, and provides assurance that allowable pool temperatures are not exceeded. The Frequencies are further justified in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

REFERENCES

1. UFSAR, Section 6.2.1.3.

2. ~~NEDC-22170, Dresden 2 and 3 Nuclear Generating Plant Suppression Pool Temperature Response, July 1982.~~

3. ~~NUREG-0783.~~

4. ~~Dresden Nuclear Power Station Units 2 and 3 Plant Unique Analysis Report, COM-02-041, May 1983.~~

B 3.7 PLANT SYSTEMS

B 3.7.7 Main Turbine Bypass System

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass
33.5% capacity of the system is ~~40%~~ of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of a nine valve manifold connected to the main steam lines between the main steam isolation valves and the main turbine stop valves. Each of the nine valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Turbine Electrohydraulic Control System, as discussed in the UFSAR, Section 7.7.4 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves sequentially. When the bypass valves open, the steam flows from the main steam equalizing header to the bypass manifold through the bypass valve, to its bypass line, where an orifice further reduces the steam pressure before the steam enters the condenser.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during the turbine trip, turbine generator load rejection and feedwater controller failure transients, as discussed in the UFSAR, Sections 15.2.3.2, 15.2.2.2, and 15.1.2 (Refs. 2, 3, and 4, respectively). Opening the bypass valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in an MCPR penalty.

The Main Turbine Bypass System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

3. A required system redundant to support system(s) for the supported systems described in b.1 and b.2 above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Primary Containment Leakage Rate Testing Program

a. This program shall establish the leakage testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemption. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

b. The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 48 psig.
43.9

c. The maximum allowable primary containment leakage rate, L_a, at P_a, is 1.6% of primary containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Primary containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria is the overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

e. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3. The LHGR for Specification 3.2.3.

~~4. The LHGR and transient linear heat generation rate limit for Specification 3.2.4.~~

4~~5~~. Control Rod Block Instrumentation Setpoint for the Rod Block Monitor—Upscale Function Allowable Value for Specification 3.3.2.1.

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

1. ANF-1125(P)(A), "Critical Power Correlation - ANFB."

2. ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors."

3. XN-NF-79-71(P)(A), "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors."

4. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors."

5. XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump Boiling Water Reactors Reload Fuel."

6. ANF-913(P)(A), "CONTRANSA2: A Computer Program for Boiling Water Reactor Transient Analysis."

7. XN-NF-82-06(P)(A), Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1 Extended Burnup Qualification of ENC 9x9 BWR Fuel.

8. ANF-89-14(P)(A), Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advance Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel.

(continued)

ATTACHMENT C

Proposed Changes to Operating Licenses and Technical Specifications for Dresden Nuclear Power Station, Units 2 and 3

INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

According to 10 CFR 50.92(c), "Issuance of Amendment," a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

Involve a significant increase in the probability or consequences of an accident previously evaluated; or

Create the possibility of a new or different kind of accident from any accident previously evaluated; or

Involve a significant reduction in a margin of safety.

In support of this determination, an evaluation of each of the three criteria set forth in 10 CFR 50.92 is provided below regarding the proposed license amendment.

Overview

Commonwealth Edison (ComEd) Company is requesting changes to Facility Operating License Nos. DPR-19 and DPR-25, and Appendix A, Technical Specifications (TS), for Dresden Nuclear Power Station (DNPS), Units 2 and 3. The proposed changes will revise the maximum power level specified in each unit's license, and TS definition of rated thermal power. In addition, other TS changes associated with this power uprate request are proposed. The specific changes requested are as follows.

- The maximum power level specified in each unit's license will be increased.
- The allowance to credit containment overpressure to assure adequate Net Positive Suction Head (NPSH) for the Emergency Core Cooling System (ECCS) pumps during a design basis accident (DBA) will be revised.
- The value of Rated Thermal Power (RTP) in the definitions will be increased.
- The definition of the Fuel Design Limiting Ratio for Centerline Melt (FDLRC) will be deleted.
- The specification for the Average Power Range Monitor (APRM) gain and setpoint adjustment will be deleted as a result of the implementation of the APRM/Rod Block Monitor (RBM) TS (ARTS) power and flow dependent limits.
- Reactor Protection System (RPS) instrumentation changes will be implemented.
- The maximum allowable time delay associated with the Isolation Condenser (IC) instrumentation will be reduced.
- Primary Containment Isolation instrumentation changes will be implemented.
- The peak calculated containment internal pressure P_a, for the design basis loss of coolant accident (LOCA) will be updated.
- The requirement to include the Transient Linear Heat Generation Rate (TLHGR) in the Core Operating Limits Report (COLR) will be deleted as a result of the implementation of the ARTS power and flow dependent limits.

The DNPS has completed comprehensive extended power uprate (EPU) analyses to increase the licensed reactor power level from 2527 Megawatts-thermal (MWt) to 2957 MWt for both Units 2 and 3. The EPU program included a reanalysis or evaluation of DBAs, non-LOCA accidents, Nuclear Steam Supply System (NSSS) and balance of plant (BOP) structures, systems and components. Major NSSS and BOP components and systems have been assessed with respect to the bounding conditions expected for operation at the uprated power level. The results of the analyses and evaluations have yielded acceptable results and demonstrated that all design basis acceptance criteria will continue to be met during uprated power operations. The detailed analysis is presented in General Electric (GE) Report NEDC-21962P, "Safety Analysis Report for Dresden 2 & 3 Extended Power Uprate," dated December 2000.

The analyses and evaluations supporting the proposed changes directly related to power uprate were completed using the guidelines in GE Topical Report NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate." Certain issues are evaluated generically and have been submitted to the NRC in GE Topical Report NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate." The NRC has approved both of these topical reports G.L. Sozzi (GE), "Staff Position Concerning General Electric Boiling-Water Reactor Extended Power Uprate Program," dated February 8, 1996,and J.F. Quirk (GE), "Staff Safety Evaluation of General Electric Boiling Water Reactor (BWR) Extended Power Uprate Generic Analyses," dated September 14, 1998.

Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

As summarized below, the increase in power level with Average Power Range Monitor (APRM) / Rod Block Monitor (RBM) Technical Specifications (ARTS) power and flow dependent limits improvements and the related Technical Specification (TS) changes discussed herein will not significantly increase the probability or consequences of an accident previously evaluated.

The probability of design basis accidents (DBAs) occurring is not affected by the increased power level or by the ARTS power and flow dependent limits, because plant equipment still complies with the applicable regulatory and design basis criteria. An evaluation of the Boiling Water Reactor (BWR) probabilistic risk assessments concludes that the calculated core damage frequencies do not significantly change due to extended power uprate (EPU) or ARTS power and flow dependent limits. Scram setpoints are established such that there is no significant increase in scram frequency due to uprate. No new challenges to safety-related equipment result from EPU or ARTS power and flow dependent limits.

Radiological release events have been evaluated, and shown to meet the requirements of 10 CFR 100, "Reactor Site Criteria." Therefore, the changes in consequences of hypothetical accidents are insignificant. The EPU accident evaluation results do not exceed any of the NRC approved acceptance limits. The spectrum of hypothetical accidents and transients has been investigated, and are shown to meet the plant's currently licensed regulatory criteria. In the area of core design, for example, the fuel operating limits such as Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Safety Limit Minimum Critical Power Ratio (SLMCPR) are still met, and fuel reload analyses will show that plant transients meet the criteria accepted by the NRC as specified in GE Topic Report NEDO-24011, "GESTAR II." Challenges to fuel are evaluated, and shown to still meet the criteria of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors" and Appendix K, "ECCS Evaluations Models."

Challenges to the containment have been evaluated, and the containment and its associated cooling systems continue to meet 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 38, "Long Term Cooling," and Criterion 50, "Containment."

The implementation of ARTS power and flow dependent limits does not affect the radiological analysis result from any postulated accident, nor does it affect the containment analysis.

The additional TS changes directly support the increased power level. All of these changes are either administrative or are proposed to ensure that the plant response to accidents and transients remain within acceptance criteria.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

As summarized below, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Equipment that could be affected by EPU or ARTS power and flow dependent limits and the additional TS changes have been evaluated. No new operating mode, safety-related equipment lineup, accident scenario or equipment failure mode is involved with EPU. The full spectrum of accident considerations, defined in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants LWR Edition," has been evaluated, and no new or different kind of accident has been identified. EPU and ARTS power and flow dependent limits uses already developed technology, and applies it within the capabilities of already existing plant equipment in accordance with presently existing regulatory criteria. Industry experience with ARTS and BWRs with higher power levels than described herein have not identified any new power dependent or ARTS related accident.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Does the proposed change involve a significant reduction in a margin of safety?

As summarized below, these changes will not involve a significant reduction in a margin of safety.

EPU affects only design and operational margins. Challenges to the fuel, reactor coolant pressure boundary, and containment were reanalyzed for EPU conditions. The fuel integrity is maintained by meeting existing design and regulatory limits. The calculated loads of all affected structures, systems and components, including the reactor coolant pressure boundary, remain within design allowables for all DBA categories. The containment performance analysis demonstrates that the containment remains within all of its design limits following the most severe DBA.

The use of ARTS power and flow dependent limits improvements ensures that the plant does not exceed any fuel thermal limit, and thus, the margin of safety is not affected.

Because the plant reactions to transients and accidents do not result in exceeding the presently approved NRC acceptance limits, these changes do not involve a significant reduction in a margin of safety.

Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Conclusion

An EPU to 117% of original rated power with ARTS power and flow dependent limits and supporting TS changes has been investigated. The method for achieving higher power is to slightly increase some plant operating parameters. The plant licensing challenges have been evaluated and demonstrate how this uprate with ARTS power and flow dependent limits can be accommodated without a significant increase in the probability or consequences of an accident previously evaluated, without creating the possibility of a new or different kind of accident from any accident previously evaluated, and without exceeding any presently existing regulatory limits or acceptance criteria applicable to the plant which might cause a reduction in a margin of safety.

Having arrived at negative declarations with regards to the criteria of 10 CFR 50.92, this assessment concludes that power uprate of the amount described herein and ARTS power and flow dependent limits do not involve a Significant Hazards Consideration.

ATTACHMENT F

Proposed Changes to Operating Licenses and Technical Specifications for Quad Cities Nuclear Power Station, Units 1 and 2

GE AFFIDAVIT FOR NEDC-32961P

General Electric Company

AFFIDAVIT

I, **George B. Stramback,** being duly sworn, depose and state as follows:

(1) I am Project Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary report NEDC-32961P, *Safety Analysis Report for Quad Cities 1 & 2 Extended Power Uprate,* Class III (GE Proprietary Information), dated December 2000. This document, taken as a whole, constitutes a proprietary compilation of information, some of it also independently proprietary, prepared by the General Electric Company. The independently proprietary elements are identified by bars marked in the left margin adjacent to the specific material.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

Both the compilation as a whole and the marked independently proprietary elements incorporated in that compilation are considered proprietary for the reason described in items (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. That information (both the entire body of information in the form compiled in this document, and the marked individual proprietary elements) is of a sort customarily held in confidence by GE, and has, to the best of my knowledge, consistently been held in confidence by GE, has not been publicly disclosed, and is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified by bars in the margin is classified as proprietary because it contains detailed results and conclusions from these evaluations, utilizing analytical models and methods, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of transient and accident events in the GE Boiling Water Reactor ("BWR"). The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several million dollars.

The remainder of the information identified in paragraph (2), above, is classified as proprietary because it constitutes a confidential compilation of information, including detailed results of analytical models, methods, and processes, including computer codes, and conclusions from these applications, which represent, as a whole, an integrated process or approach which GE has developed, obtained NRC approval of, and applied to perform evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability of a given increase in licensed power output for a GE BWR. The development and approval of this overall approach was achieved at a significant additional cost to GE, in excess of a million dollars, over and above the very large cost of developing the underlying individual proprietary analyses.

To effect a change to the licensing basis of a plant requires a thorough evaluation of the impact of the change on all postulated accident and transient events, and all other regulatory requirements and commitments included in the plant's FSAR. The analytical process to perform and document these evaluations for a proposed power uprate was developed at a substantial investment in GE resources and expertise. The results from these evaluations identify those BWR systems and components, and those postulated events, which are impacted by the changes required to accommodate operation at increased power levels, and, just as importantly, those which are <u>not</u> so impacted, and the technical justification for not considering the latter in changing the licensing basis. The scope thus determined forms the basis for GE's offerings to support utilities in both performing analyses and providing licensing consulting services. Clearly, the scope and magnitude of effort of any attempt by a competitor to effect a similar licensing change can be narrowed considerably based upon these results. Having invested in the initial evaluations and developed the solution strategy and process described in the subject document GE derives an important competitive advantage in selling and performing these services. However, the mere knowledge of the impact on each system and component reveals the process, and provides a guide to the solution strategy.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive

physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods, including justifications for not including certain analyses in applications to change the licensing basis.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to avoid fruitless avenues, or to normalize or verify their own process, or to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions. In particular, the specific areas addressed by any document and submittal to support a change in the safety or licensing bases of the plant will clearly reveal those areas where detailed evaluations must be performed and specific analyses revised, and also, by omission, reveal those areas not so affected.

While some of the underlying analyses, and some of the gross structure of the process, may at various times have been publicly revealed, enough of both the analyses and the detailed structural framework of the process have been held in confidence that this information, in this compiled form, continues to have great competitive value to GE. This value would be lost if the information as a whole, in the context and level of detail provided in the subject GE document, were to be disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources, including that required to determine the areas that are not affected by a power uprate and are therefore blind alleys, would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing its analytical process.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

George B. Stramback, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 20th day of December 2000.

George B. Stramback

George B. Stramback
General Electric Company

Subscribed and sworn before me this 20th day of December 2000.

Vicky D Schroer

Notary Public, State of California

VICKY D. SCHROER
Commission # 1224251
Notary Public - California
Santa Clara County
My Comm. Expires Jun 12, 2003

PLANT MODIFICATIONS REQUIRED TO SUPPORT POWER UPRATE

The following presents an overview of the facility changes necessary to achieve the target electrical power output of 912 MWe.

- Various instruments will require scaling/setpoint changes.
- A modification to provide tripping of the 4th condensate pump on a LOCA will be implemented to allow the continued use of the feedwater pumps.
- A fault current limiting arrangement will be implemented to maintain non-safety bus short circuit ratings after a postulated loss of an auxiliary transformer in conjunction with a short circuit.
- A reactor recirculation pump runback on a loss of feedwater flow or the loss of a condensate pump will be implemented to reduce the potential for a scram on reactor low water level and allow continued operation.
- An additional steam line resonance compensator card designed to attenuate third order harmonics will be installed in the electro-hydraulic control system to reduce electrical noise in the system.
- A new high-pressure turbine rotor will be installed as a result of the increased steam flow associated with operation at uprated power conditions.
- Turbine cross around relief valve alterations will be performed to ensure that pressure limitations are not exceeded.
- Selected heater drain valve normal drain trim replacements will be performed due to the increase in drain flow.
- Some feedwater heater relief valves will be adjusted or replaced and the heaters will be rerated to compensate for the increased feedwater flow and the associated pressure change.
- Condenser tube staking is planned for the main condensers to provide adequate protection against tube vibration damage at uprated power conditions.
- An additional condensate demineralizer will be added to process the increased flow.
- Various support and piping modifications will be performed due to the increased temperature in torus-attached piping and increased temperature and flow in the main steam and feedwater systems.
- Restriction orifices to the stator water cooling system will be resized to accommodate the increased heat load.
- Modifications to the steam dryer will be performed to reduce moisture carryover.