Northeast
Nuclear Energy

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The Northeast Utilities System

JUL 31 2000

Docket No. 50-336
B18185

Re: 10 CFR 50.90

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

Millstone Nuclear Power Station, Unit No. 2
License Amendment Request - Unreviewed Safety Question
Proposed Revision to Final Safety Analysis Report
Fuel Centerline Melt Linear Heat Rate Limit (PLAR 2-00-2)

Northeast Nuclear Energy Company (NNECO) has determined that changing the method used to determine the Fuel Centerline Melt Linear Heat Rate Limit (FCMLHRL) involves an Unreviewed Safety Question (USQ). Therefore, per 10 CFR 50.59(c), NNECO requests that the Nuclear Regulatory Commission (NRC) review and approve the changes to the Final Safety Analysis Report (FSAR) through an amendment to Operating License DPR-65, pursuant to 10 CFR 50.90. This license amendment request deals with changes in the Millstone Unit No. 2 FSAR due to changing the method used to determine the FCMLHRL.

The proposed license amendment request will affect several sections of the FSAR. Changing the method used to determine the FCMLHRL may result in a reduction in the margin of safety as defined in the basis of Technical Specifications. Therefore, these changes are deemed to involve an USQ according to 10 CFR 50.59 (a)(2)(i).

Attachment 1 provides a discussion of the proposed changes and the safety summary. Attachment 2 provides the Significant Hazards Consideration. Attachment 3 provides the FSAR pages with the changes indicated. Changing the method used to determine the FCMLHRL impacts also the Millstone Unit No. 2 Technical Specification Bases, Section 2.1.1, "Reactor Core." A change to the Technical Specification Bases to

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remove reference to the limit of 21kW/ft will be implemented upon the NRC approval of this proposed license amendment request. The proposed Bases change is contained in Attachment 4.

Environmental Considerations

NNECO has reviewed the proposed license amendment request against the criteria of 10 CFR 51.22 for environmental considerations. The proposed changes will not significantly increase the type and amounts of effluents that may be released offsite. In addition, this amendment request will not significantly increase individual or cumulative occupational radiation exposures. Therefore, NNECO has determined the proposed changes will not have a significant effect on the quality of the human environment.

Conclusions

The proposed changes do not involve a significant impact on public health and safety (see the Safety Summary provided in Attachment 1) and do not involve a Significant Hazards Consideration pursuant to the provisions of 10 CFR 50.92 (see the Significant Hazards Consideration provided in Attachment 2).

Plant Operations Review Committee and Nuclear Safety Assessment Board

The Plant Operations Review Committee and Nuclear Safety Assessment Board have reviewed and concurred with the determinations.

Schedule

We request issuance of this amendment for Millstone Unit No. 2 by March 31, 2001, with the amendment to be implemented within 60 days of issuance.

State Notification

In accordance with 10 CFR 50.91(b), a copy of this license amendment request is being provided to the State of Connecticut.

There are no regulatory commitments contained within this letter.

If you should have any questions on the above, please contact Mr. Ravi Joshi at (860) 440-2080.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

Necci

Raymond P. Necci
Vice President - Nuclear Technical Services

Subscribed and sworn to before me

this 31st day of July, 2000

Sandra J. Anton
Notary Public

Date Commission Expires:

SANDRA J. ANTON
NOTARY PUBLIC
COMMISSION EXPIRES
MAY 31, 2005

Attachments (4)

cc: H. J. Miller, Region I Administrator
J. I. Zimmerman, NRC Project Manager, Millstone Unit No. 2
S. R. Jones, Senior Resident Inspector, Millstone Unit No. 2

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
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Attachment 1

Millstone Nuclear Power Station, Unit No. 2

License Amendment Request - Unreviewed Safety Question
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Discussion of Changes

License Amendment Request - Unreviewed Safety Question
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Introduction and Background

The Reactor Protection System (RPS) relies on input parameters, which are denoted as Limiting Safety System Settings (LSSS). The values or functional representation of the LSSSs are calculated to ensure adherence to the Specified Acceptance Fuel Design Limits (SAFDL) during steady state and Anticipated Operational Occurrences (AOO).

The SAFDLs are limits on the fuel and cladding established in order to preclude fuel failure. These limits may not be exceeded during steady-state operation or during AOOs. The SAFDLs are used to establish the reactor setpoints to ensure safe operation of the reactor. One of the specific SAFDLs used to establish the setpoints is the Local Power Density (LPD) which coincides with the fuel centerline melt. The LPD limit for Millstone Unit No. 2 is the Fuel Centerline Melt Linear Heat Rate Limit (FCMLHRL).

Northeast Nuclear Energy Company (NNECO) is proposing a change in the method used to determine the limiting FCMLHRL. The proposed change represents a departure from the use of the fixed value of 21 kW/ft for the FCMLHRL, which is being used in the current cycle, to a value that will be calculated on a cycle by cycle basis using the Siemens Power Corporation (SPC) approved methodology. This methodology was reviewed and approved by the Nuclear Regulatory Commission (NRC) and is documented in SPC report XN-NF-82-06(P)(A).[(1)] The SPC report is listed as document No. 11 in section 6.9.1.8.b of the Millstone Unit No. 2 Technical Specifications describing the previously reviewed and approved analytical methods used to determine the core operating limits. The change in the method of setting the limiting FCMLHRL value is safe. However, it may result in a reduction in the margin of safety. Therefore, this change is deemed to involve an Unreviewed Safety Question (USQ). Therefore, per 10 CFR 50.59(c), NNECO requests that the NRC review and approve the changes to the Final Safety Analysis Report (FSAR) through an amendment to Operating License DPR-65, pursuant to 10 CFR 50.90.

The proposed FSAR changes which will require NRC approval are contained in Sections 3.5.2.1.2, "Fuel Pellet Temperatures," 3.5, "REFERENCES," 7.2.3.3.10, "High Local Power Density Trip," 14.0.7.2, "Specified Acceptable Fuel Design Limits," 14.0.7.3.1, "Local Power Density," 14.0.11, "Plant Licensing Basis and Single Failure Criteria," 14.0, "REFERENCES," 14.1.3.6, "Analysis Results," 14.1.3.7, "Conclusion,"

[(1)] "Qualification of Exxon Nuclear Fuel for Extended Burnup," XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4 and 5, Exxon Nuclear Company, October 1986.

14.1.5.1.6.1.3, "Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results," 14.1.5.1.6.2.3, "Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results," 14.1.5.1.7, "Conclusions," 14.1.5.2.6.1.5, "Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results," 14.1.5.2.6.2.5, "Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results," 14.1.5.2.7, "Conclusions," 14.1.5.3, "Radiological Consequences of a Main Steam Line Break," 14.2.1.6, "Analysis Results," 14.2.1.7, "Conclusion," 14.2.4.7, "Conclusion," 14.3.1.6, "Analysis Results," 14.3.1.7, "Conclusion," 14.3.3.6, "Analysis Results," 14.3.3.7, "Conclusion," 14.4.2.6, "Analysis Results," 14.4.2.7, "Conclusion," 14.6.1.6, "Analysis Results," and 14.6.1.7, "Conclusions;" and Tables 7.2-1, "Reactor Trip and Pretrip Set Points," 14.1.5.2-6, "Post-Scram Steam Line Break Analysis Summary," 14.1.5.3-1, "Assumptions Used in Main Steam Line Break Analysis," and 14.1.5.3-2, "Summary of Millstone Unit 2 MSLB Accident Doses."

Current Licensing Bases

The primary thermal-hydraulic design criteria for SPC reload fuel assure that fuel rod integrity is maintained during normal operation and AOOs. One of these criteria is that fuel centerline temperatures remain below the melting point of the fuel pellets. Observance of this criterion ensures that AOOs do not result in fuel rod failures or loss of functional capability. A conservative FCMLHRL of 21 kW/ft is used for Millstone Unit No. 2. The maximum Linear Heat Rate (LHR) for normal operation and anticipated transients is typically well below the conservative value of 21 kW/ft. Safety of the core is ensured by:

1. Proper setting of the LPD trip limit. If the LPD limit of 21 kW/ft is exceeded, the RPS will produce a reactor trip.
2. One of the requirements of 10 CFR 50, Appendix A, Criteria 10, 20, 25 and 29 is that the design and operation of the plant and the RPS assure that SAFDLs not be exceeded during AOOs. One of the SAFDLs is that the fuel shall not experience centerline melt by limiting FCMLHRL to 21 kW/ft.
3. The peak LHR limit is used to determine fuel failures for the chapter 14 accident analyses.

Proposed Changes in Licensing Bases:

NNECO is proposing a change in the method used to determine the limiting FCMLHRL. The proposed change represents a departure from the use of the fixed value of 21 kW/ft for the FCMLHRL, which is being used in the current cycle, to a value that will be calculated on a cycle by cycle basis using the SPC approved methodology. This methodology was reviewed and approved by the NRC and is documented in SPC report XN-NF-82-06(P)(A). The SPC report is a part of the document list in section 6.9.1.8.b of the Millstone Unit No. 2 Technical Specifications, which describes the previously reviewed and approved analytical methods used to determine the core operating limits. The proposed methodology change will be implemented upon NRC approval.

Description of Proposed FSAR Changes

NNECO requests that the NRC review and approve, through an amendment to Operating License DPR-65 pursuant to 10 CFR 50.90, the proposed FSAR changes which affect Sections 3.5.2.1.2, 3.5, 7.2.3.3.10, 14.0.7.2, 14.0.7.3.1, 14.0.11, 14.0, 14.1.3.6, 14.1.3.7, 14.1.5.1.6.1.3, 14.1.5.1.6.2.3, 14.1.5.1.7, 14.1.5.2.6.1.5, 14.1.5.2.6.2.5, 14.1.5.2.7, 14.1.5.3, 14.2.1.6, 14.2.1.7, 14.2.4.7, 14.3.1.6, 14.3.1.7, 14.3.3.6, 14.3.3.7, 14.4.2.6, 14.4.2.7, 14.6.1.6, and 14.6.1.7; and Tables 7.2-1, 14.1.5.2-6, 14.1.5.3-1, and 14.1.5.3-2. The proposed changes in the FSAR will remove reference to the FCMLHRL of 21kw/ft and replace it with a reference to the SPC approved methodology as described in SPC report XN-NF-82-06(P)(A).

Safety Summary

This license amendment request deals with changes in the Millstone Unit No. 2 FSAR due to changing the method used to determine the FCMLHRL. The proposed change represents a departure from the use of the fixed value of 21 kW/ft for the FCMLHRL, which is being used in the current cycle, to a value that will be calculated on a cycle by cycle basis using the SPC approved methodology. This methodology was reviewed and approved by the NRC and is documented in SPC report XN-NF-82-06(P)(A).

Millstone Unit No. 2 currently implements a conservative FCMLHRL of 21 kW/ft. With SPC's setpoint verification methodology, the value of the FCMLHRL that can be supported is determined for each reload cycle. The value of the FCMLHRL is verified for each reload, but does not typically change significantly between cycles. This limit is determined for a standard fuel rod. The current enrichment cutbacks in the gadolinia bearing rods limit their relative power such that the maximum FCMLHRL for a gadolinia bearing fuel rod will be sufficiently below the standard fuel rods to prevent centerline melt. In future application of this methodology, the peak Linear Heat Rates (LHR) calculated from transient analyses will be compared to the FCMLHRL for the cycle. The LPD LSSS verification analysis for future applications will use the cycle dependent FCMLHRL.

The departure from using a standard value of 21 kW/ft to the use of a cycle specific calculated value represents in itself a change in the method of setting the limiting FCMLHRL value. The change in the method of setting the limiting FCMLHRL value may result in a reduction in the margin of safety. However, the proposed changes are safe because SPC has justified, using NRC generically approved methodology, that with a higher value of the FCMLHRL the fuel will not experience centerline melt. In other words, a higher FCMLHRL may allow a higher fuel temperature but will continue to protect fuel against centerline melt. Therefore, it can be concluded that these changes are safe even though the proposed changes represent an USQ.

Attachment 2

Millstone Nuclear Power Station, Unit No. 2

License Amendment Request - Unreviewed Safety Question
Proposed Revision to Final Safety Analysis Report
Fuel Centerline Melt Linear Heat Rate Limit (PLAR 2-00-2)
Significant Hazards Consideration

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Significant Hazards Consideration

Significant Hazards Consideration

In accordance with 10 CFR 50.92, Northeast Nuclear Energy Company (NNECO) has reviewed the proposed changes and has concluded that they do not involve a Significant Hazards Consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

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

   This license amendment request deals with changes in the Millstone Unit No. 2 Final Safety Analysis Report (FSAR) due to changing the method used to determine the FCMLHRL. The proposed change represents a departure from the use of the fixed value of 21 kW/ft for the FCMLHRL, which is being used in the current cycle, to a value that will be calculated on a cycle by cycle basis using the SPC approved methodology. This methodology was reviewed and approved by the Nuclear Regulatory Commission (NRC) and is documented in Siemens Power Corporation (SPC) report XN-NF-82-06(P)(A).[1] The value of the FCMLHRL is verified for each reload, but does not typically change significantly between cycles. This limit is determined for a standard fuel rod. The current enrichment cutbacks in the gadolinia bearing rods limit their relative power such that the maximum FCMLHRL for a gadolinia bearing fuel rod will be sufficiently below the standard fuel rods to prevent centerline melt. In future applications of this methodology, the peak Linear Heat Rates (LHR) calculated from transient analyses will be compared to the FCMLHRL for the cycle. The Local Power Density (LPD) Limiting Safety System Settings (LSSS) verification analysis for future applications will use the cycle dependent FCMLHRL. Therefore, It can be concluded that these FSAR changes are safe and that the cycle specific calculated FCMLHRL has no impact on plant equipment operation. Further more, the change in the method of determining the FCMLHRL only impacts the analytical determination of failed fuel and has no direct impact on the accident scenario. Accordingly, this change cannot affect the likelihood of these events. Therefore, the proposed changes will not increase the probability of occurrence of accidents previously evaluated.

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[1] "Qualification of Exxon Nuclear Fuel for Extended Burnup," XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4 and 5, Exxon Nuclear Company, October 1986.

The change in the method of determining the FCMLHRL will continue to conservatively estimate fuel failures. Since the proposed FSAR changes will have no impact on the analysis of the events, they cannot affect the likelihood or consequences of these events. Therefore, the proposed FSAR changes will not increase the consequences of accidents previously evaluated.

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

   The proposed FSAR changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. The FSAR changes do not introduce any new failure modes. Therefore, the changes will not increase the probability of a new or different kind of accident from any accidents previously evaluated.

3. Involve a significant reduction in a margin of safety.

   The purpose of the proposed changes is to document a change in the method used to determine FCMLHRL in the Millstone Unit No. 2 FSAR. The change in methodology may result in a FCMLHRL that is higher than the previous limit of 21 kW/ft. Therefore, the proposed changes may lead to a reduction of the margin of safety. However, the proposed changes are safe because SPC has justified, using NRC generically approved methodology, that with a higher value of the FCMLHRL the fuel will not experience centerline melt. In other words, a higher FCMLHRL may allow a higher fuel temperature but will continue to protect fuel against centerline melt. Therefore, it can be concluded that the FSAR changes are safe and do not significantly reduce the margin of safety.

As described above, this license amendment request does not involve a significant increase in the probability of an accident previously evaluated, does not involve a significant increase in the consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident from any accident previously evaluated, and does not result in a significant reduction in a margin of safety. Therefore, NNECO has concluded that the proposed changes do not involve a SHC.

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Attachment 3

Millstone Nuclear Power Station, Unit No. 2

License Amendment Request - Unreviewed Safety Question
Proposed Revision to Final Safety Analysis Report
Fuel Centerline Melt Linear Heat Rate Limit (PLAR 2-00-2)
Marked Up Pages

List of Pages Affected

Section / Subsection Titles	Section Numbers	Page Numbers
Fuel Pellet Temperatures	3.5.2.1.2	3.5-2
REFERENCES	3.5	3.5-16
High Local Power Density Trip	7.2.3.3.10	7.2-12
Specified Acceptable Fuel Design Limits	14.0.7.2	14.0-5
Local Power Density	14.0.7.3.1	14.0-6
Plant Licensing Basis and Single Failure Criteria	14.0.11	14.0-10
REFERENCES	14.0	14.0-12
Analysis Results	14.1.3.6	14.1-5
Conclusion	14.1.3.7	14.1-5
Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results	14.1.5.1.6.1.3	14.1-13
Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results	14.1.5.1.6.2.3	14.1-14
Conclusions	14.1.5.1.7	14.1-14
Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results	14.1.5.2.6.1.5	14.1-23
Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results	14.1.5.2.6.2.5	14.1-24
Conclusions	14.1.5.2.7	14.1-25
Radiological Consequences of a Main Steam Line Break	14.1.5.3	14.1-25
Radiological Consequences of a Main Steam Line Break	14.1.5.3	14.1-26
Analysis Results	14.2.1.6	14.2-3
Conclusion	14.2.1.7	14.2-3
Conclusion	14.2.4.7	14.2-7
Analysis Results	14.3.1.6	14.3-2
Conclusion	14.3.1.7	14.3-3
Analysis Results	14.3.3.6	14.3-4
Conclusion	14.3.3.7	14.3-4
Analysis Results	14.4.2.6	14.4-4
Conclusion	14.4.2.7	14.4-4
Analysis Results	14.6.1.6	14.6-2
Conclusions	14.6.1.7	14.6-2

Table Titles	Table Numbers	Page Numbers
Reactor Trip and Pretrip Set Points	7.2-1	1 of 1
Post-Scram Steam Line Break Analysis Summary	14.1.5.2-6	1 of 1
Assumptions Used in Main Steam Line Break Analysis	14.1.5.3-1	1 of 1
Summary of Millstone Unit 2 MSLB Accident Doses	14.1.5.3-2	1 of 1

Figure Titles	Figure Numbers	Page Numbers
N/A	N/A	N/A

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To ensure that sufficient coolant flow reaches the fuel, the amount of coolant flow which bypasses the core through the guide tubes must not excessively reduce the active core flow. The guide tube coolant flow must, however, be sufficient to ensure that coolant in the guide tubes will not boil and ensure adequate cooling of the CEA fingers. The CEA drop time in the guide tubes must also meet the criterion of 90 percent insertion within 2.75 seconds to ensure that scram performance is in accordance with plant Technical Specifications.

Although the coolant velocity, its distribution, and the coolant voids affect the thermal margin, design limits need not be applied to these parameters because they are not themselves limiting with respect to thermal margin. These parameters are included in the thermal margin analyses and thus affect the thermal margin to the design limits.

3.5.2 Thermal and Hydraulic Characteristics of the Design

3.5.2.1 Fuel Temperatures

The ROXDEX2 code (Reference 3.5-1) incorporates models to describe the thermal and mechanical behavior of the fuel rod in a flow channel including the gas release, swelling, densification, and cracking in the pellet; the gap conductance; the radial thermal conduction; the free volume and gas pressure internal to the fuel rod; the fuel and cladding deformations; and the cladding corrosion as a function of burnup. The calculations are performed on a time-incremental basis with conditions being updated at each calculated increment.

3.5.2.1.1 Fuel Cladding Temperatures

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The RODEX2 thermal-hydraulic model (Reference 3.5-1) calculates the lowest cladding surface temperature based on one of two heat transfer regimes; i.e., forced convection and fully developed nucleate boiling. The forced convection and fully developed nucleate boiling heat transfer correlations in RODEX2 were developed by Kays and Thom et al., respectively.

3.5.2.1.2 Fuel Pellet Temperatures

The RODEX2 radial temperature distribution model begins with the standard differential equation of heat conduction (Poisson Equation) for an isotropic solid with internal heat generation. The equation is written in cylindrical coordinates assuming that the thermal conductivity of the fuel is a function of fuel temperature, but is independent of position. With additional assumptions of axial symmetry, negligible heat conduction in the axial direction, and steady-state conditions, a one-dimensional (i.e., radial) steady-state form of the equation is derived and employed.

defined as the Fuel Centerline Melt Linear Heat Rate (FCMLHR) limit and is expressed in

The minimum power level required to produce centerline melt in Zircaloy clad uranium fuel rods is typically about 23 kW/ft. A conservative ~~centerline melt linear heat generation rate (LHGR)~~ limit of 21 kW/ft is used for Millstone Unit 2. The maximum LHGR for normal operation and anticipated transients is typically well below the conservative ~~criterion of 21 kW/ft~~ FCM LHR limit. It should be noted that a gadolinia-bearing fuel rod will, for a given LHGR, operate with a higher fuel temperature than an all-uranium-bearing fuel rod. However, the U-235 enrichment in gadolinia bearing fuel rods is typically low such that gadolinia rods are

Cycle specific FCMLHR

This FCMLHR is determined using the methodology of Reference 3.5-22.

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3.5-15 H. S. Kao, C. D. Morgan, and W. B. Parker, "Prediction of Flow Oscillation in Reactor Core Channel," Trans. ANS Vol. 16, pp. 212-213 (1973).

3.5-16 A. E. Bergles and M. Suo, "Investigation of Boiling Water Flow Regimes at High Pressure," Dynatech Corp. NYO-3304-8 (February 1966).

3.5-17 E. R. Hosler, "Flow Patterns in High Pressure Two-Phase (Steam-Water) Flow with Heat Addition," 9th National Heat Transfer Conf., Chemical Engineering Progress Symposium Series, No. 82, Vol. 64, pp. 54-66 (August 1967).

3.5-18 Weisman et. al., "Experimental Determination of the Departure from Nucleate Boiling in Large Rod Bundles at High Pressure," 9th National Heat Transfer Conf., Chemical Engineering Progress Symposium Series, No. 82, Vol. 64, pp. 114-125 (August 1967).

3.5-19 Siemens Power Corporation Calculation Notebook, E-5188-011-01, "Millstone Unit 2 Data Requirements for Reload Analysis," September 1987.

3.5-20 Letter, R. I. Wescott (SPC) to C. H. Wu (NU), "Transmittal of Bases for New Uncertainties in the Setpoint Analysis for Millstone Unit 2," RIW:97:049, February 27, 1998. 99-70

3.5-21 ANSI/ANS-19.6.1 "Reload Startup Physics Tests for Pressurized Water Reactors," 1985.

3.5-22 "Qualification of Exxon Nuclear Fuel for Extended Burnup," XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4 and 5, Exxon Nuclear Company, October 1986.

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This FCMLHR is determined using the methodology of XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4 and 5 ("Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company, October 1986.)

Since credit has not been taken for the equipment protective trips in the Safety Analysis of the plant, they do not fall within the scope of IEEE 279. In the case of the Millstone Unit No. 2 RPS, the design criteria listed in Section 7.2.1.2 (which includes IEEE 279) apply to all trip functions including the equipment protective trips.

7.2.3.3.9 High Containment Pressure

A trip is provided on high containment pressure in order to assure that the reactor is tripped concurrent with safety injection actuation.

Four pressure measurement channels provide analog signals to bistable trip units which are connected in a two-out-of-four coincidence logic to initiate the protective action if the containment pressure exceeds a preselected value.

7.2.3.3.10 High Local Power Density Trip

The minimum power level required to produce centerline melt in Zircaloy clad uranium fuel rods is defined as the Fuel Centerline Melt Linear Heat Rate (FCMLHR) limit and is expressed in KW/ft.

The high LPD trip is provided to prevent the peak LPD in the fuel from exceeding the FCMLHR Limit ~~21 KW/ft~~ during anticipated operational occurrences thereby assuring that the melting point of the UO_2 fuel will not be reached. ~~A value of 21 KW/ft is well below the value corresponding to fuel centerline melting.~~ (5/90)

A reactor trip is initiated whenever the axial offset exceeds either a high or low calculated setpoint as described below. The axial offset is calculated from upper and lower ex-core neutron detector channels. The calculated setpoints are generated as a function of the core power level with the CEA group position being inferred from the core power. The trip is automatically bypassed below 15 percent power.

Consistent with the Technical Specifications, the maximum azimuthal tilt and the maximum CEA deviation permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with Technical Specification is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a High Power Level Trip is assumed.

Figure 7.2-13 shows a block diagram of a typical channel. Circuits in the Power Range Safety Channel generate signals proportional to the sum of and the difference between the upper and lower detector outputs. An axial offset signal is formed as a linear function of the ratio of the difference to the sum and compared with upper and lower limits generated from a modified power signal described in Section 7.2.3.3.8. The offset signal is also used in the Thermal Margin Trip as previously described.

If the axial offset exceeds either calculated limit, a contact in the calculator opens and deenergizes the trip relays in an auxiliary trip unit. The pretrip relay is similarly released if a narrower envelope is exceeded. Trip also occurs if either "trip test" knob on the Power Range Safety Channel is moved off the zero position.

7.2.3.3.11 Manual Trip

A manual reactor trip is provided to permit the operator to trip the reactor. Depressing two pushbutton switches on the control panel causes interruption of the AC power to the CEDM power supplies. The manual trip function is testable during reactor operation.

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TABLE 7.2-1

REACTOR TRIP AND PRETRIP SET POINTS

No.	Reactor Trip	Pretrip Alarm Set Point	Trip Set Point	
1.	Reactor Coolant Pump Underspeed**	N.A.	≥830 rpm	
2.	High Power Level	2% below trip setpoint	≤10% above measured power Q	
3.	Low Reactor Coolant Flow** 4-Pump Operation, %	N.A.	≥91.7	98-73
4.	Low Steam Generator Water Level, % (Auctioneered low of SG #1, SG #2)	54	≥48.5	98-184
5.	Low Steam Generator Pressure***, psia (Auctioneered low of SG #1, SG #2)	780	≥691	98-73
6.	High Pressurizer Pressure, psia	2350	≤2397	98-73
7.	Thermal Margin/Low-Pressure**	75 psia above trip set point	Variable trip set point with minimum of 1865 psia	98-184
8.	Loss of Turbine**** (Low Hydraulic Fluid Pressure) psig	N.A.	≥500	
9.	High Containment Pressure, psig	N.A.	≤4.42	98-73
10.	Manual Trip (Push Buttons)	N.A.	N.A.	
~~11.~~	~~Local Power Density~~	~~20 kW/ft~~ < FCM LHR	~~21 kW/ft~~ FCM LHR	

**Manual inhibit permitted below 10^{-4} percent power: automatically removed above 10^{-4} percent power.

***Manual inhibit permitted below 800 psia: automatically removed above 800 psia. 98-73

****Inhibited below 15% power.

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14.0.7 Trip Setpoint Verification

Operating limits for the Millstone Unit 2 nuclear plant are summarized below. Methods of analysis for determining or verifying the operating limits are detailed in Subsection 14.0.7.5 and Reference 14.0-4. Axial power distributions and other core neutronics related parameters used in the setpoint verification analyses were generated with Siemens Power Corporation's (SPC) approved core simulator code XTGPWR (Reference 14.0-5). This data was generated on a three-dimensional core basis, as described in References 14.0-6 and 14.0-7. With this methodology, the values of F_Q used in the setpoint verification are calculated directly with a three-dimensional model and since operation within the technical specification on F_r^T limits F_Q, the need for an F_{xy}^T limit is eliminated. 99-3

Results of the analyses indicate that operating limits established for Millstone Unit 2 are acceptable.

14.0.7.1 Reactor Protection System

The reactor protection system (RPS) is designed to assure that the reactor is operated in a safe and conservative manner. The input parameters for the RPS are denoted as limiting safety system settings (LSSS). The values or functional representation of the LSSSs are calculated to ensure adherence to the specified acceptable fuel design limits (SAFDL) during steady state and anticipated operational occurrences (AOO). The safe operation of the reactor is also maintained by restricting reactor operation to conform with the LCOs, which are administratively applied at the reactor plant. The LSSS and LCO parametric values are presented in the following sections. 99-3

14.0.7.2 Specified Acceptable Fuel Design Limits

The SAFDLs are limits on the fuel and cladding established in order to preclude fuel failure. These limits may not be exceeded during steady-state operation or during AOOs. The SAFDLs are used to establish the reactor setpoints to ensure safe operation of the reactor. The specific SAFDLs used to establish the setpoints are:

(1) The local power density (LPD) which coincides with fuel centerline melt.

(2) The minimum departure from nucleate boiling ratio (MDNBR) corresponding to the accepted criterion which protects against the occurrence of departure from nucleate boiling (DNB). 99-3

The LPD limit for Millstone Unit 2 is ~~21 kW/ft~~ the FCM LHR limit. It is noted that reload the fuel may contain gadolinia-bearing fuel rods which, for a given LPD, will operate with a higher fuel temperature and will consequently have a lower LPD limit. The neutronics design of the gadolinia-bearing fuel rods is such that the maximum LPD in the gadolinia-bearing fuel rod with a standard fuel rod at the 21 kW/ft limit will be sufficiently below 21 kW/ft to prevent centerline melt. Therefore, the gadolinia-bearing fuel would not become limiting and the 21 kW/ft design limit would remain applicable. FCM LHR

The XNB critical heat flux correlation (Reference 14.0-8) was used in the thermal margin analysis with statistical parameters corresponding to an upper 95/95 value of 1.17 which is conservative relative to the 95/95 limit for XNB. Observance of the LCO will protect

The minimum power level required to produce centerline melt in Zircaloy clad uranium fuel rods is defined as the Fuel Centerline Melt Linear Heat Rate (FCM LHR) limit and is expressed in kW/ft. This FCM LHR is determined using the methodology of Reference 14.0-15.

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against DNB with 95% probability at a 95% confidence level during an AOO. The XNB correlation was qualified for application to 14x14 fuel for Combustion Engineering (CE) reactors in Reference 14.0-9. 99-3

14.0.7.3 Limiting Safety System Settings

14.0.7.3.1 Local Power Density

The LPD trip limit is the locus of the limiting values of FCMLHR core power level versus ASI that will produce a reactor trip to prevent exceeding the 21 kW/ft LPD limit. The correlation between allowed core power level and peripheral ASI was determined using methods which take into account the total calculated nuclear peaking and the measurement and calculational uncertainties associated with power peaking. The LPD barn for operation at 2700 MWt is shown in Figure 14.0.7-1 as a locus of power and ASI pairs which conservatively bounds the calculated power and ASI pairs. ASI is defined as the difference between the core power in the bottom half of the core and the top half divided by the sum of the top and bottom halves. 99-3

14.0.7.3.2 Thermal Margin/Low Pressure

The thermal margin/low pressure (TM/LP) trip protects against the occurrence of DNB during steady state operations and for many, but not all, AOOs. This reactor trip system monitors primary system pressure, core inlet temperature, core power and ASI. A reactor trip occurs when primary system pressure falls below the computed limiting core pressure, P_{var}. A statistical setpoint methodology (Reference 14.0-4) is used to verify the adequacy of the existing TM/LP trip. The methodology for the TM/LP trip accounts for uncertainties in core operating conditions, XNB DNB correlation uncertainties, and uncertainties in power peaking. The existing TM/LP trip function is given by:

$$P_{var} = 2215 \times A1\,(ASI) \times QR1\,(Q) + 14.28 \times T_{in} - 8240,$$

where Q is the higher of the thermal power and the nuclear flux power, T_{in} is the inlet temperature in °F and A1 and QR1 are shown in Figures 14.0.7-2 and 14.0.7-3, respectively.

14.0.7.3.3 Additional Trip Functions

In addition to the LPD and TM/LP trip functions, other reactor system trips have been determined to provide adherence to reactor system design criteria. The analytical setpoints for these trips are shown in Table 14.0.7-1. 99-3

14.0.7.4 Limiting Conditions for Operation

14.0.7.4.1 Departure From Nucleate Boiling 99-3

The validity of the existing LCO for allowable core power as a function of ASI was verified to ensure adherence to the SAFDL on DNB during a postulated loss-of-flow operational occurrence. The statistical analysis accounted for the effects of uncertainties associated with core operating parameters, the XNB critical heat flux correlation, and power peaking. 99-3

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The requirements of 10 CFR 50, Appendix A, Criteria 10, 20, 25 and 29 require that the design and operation of the plant and the RPS assure that the SAFDLs not be exceeded during AOOs. As per the definition of AOO in 10 CFR 50, Appendix A, "Anticipated Operational Occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power." The SAFDLs are that: 1) the fuel shall not experience centerline melt (21 kW/ft); and 2) the DNBR shall have a minimum allowable limit such that there is a 95% probability with a 95% confidence interval that DNB has not occurred (DNBR of 1.17 as calculated using the XNB DNB correlation). 99-3

14.0.12 Plot Variable Nomenclature

Some of the plotted results presented in Sections 14.1 through 14.6, use PTSPWR2 (Reference 14.0-13) and SLOTRAX (Reference 14.0-14) output variable nomenclature. Specific variables plotted are listed and defined in Table 14.0.12-1. 99-3

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14.0-14 "SLOTRAX-ML: A Computer Code for Analysis of Slow Transients in PWRs," XN-NF-85-24(A), Exxon Nuclear Company, Richland, WA 99352, September 1986.

14.0-15 "Qualification of Exxon Nuclear Fuel for Extended Burnup," XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4 and 5, Exxon Nuclear Company, October 1986.

at EOC conditions when the MTC is at its maximum negative value. Therefore, the technical specification most negative MTC limit (-28 pcm/°F) was used.

Only full-power cooldown and low-power events which credit power-dependent reactor trips have the potential to be adversely affected by power decalibration. Power decalibration is caused by density-induced changes in the reactor vessel downcomer shadowing the power-range ex-core detectors during heatup or cooldown transients. The nuclear-power levels indicated by those instruments are lower than the actual reactor power levels when the coolant entering the reactor vessel is cooler than the normal temperature for full-power operation. The Variable Overpower trip, the TM/LP trip function, and the LPD trip all depend on the indicated nuclear power level. The power decalibration effect was included in the modeling of any power-dependent reactor trips credited in this analysis. 98-182 12/98

The initial conditions for the Increase in Steam Flow event is summarized in Table 14.1.3-3.

14.1.3.6 Analysis Results

(5/90)

The transient for the limiting case (51.8% excess steam flow) is initiated by a failure which causes the steam dump to bypass valves and the turbine bypass valves to open fully. The turbine control valves are also modeled as opening fully at initiation. The increased steam flow (see Figure 14.1.3-7) creates a mismatch between the core heat generation rate and the steam generator heat removal rate. This power mismatch causes the primary-to-secondary heat transfer rate to increase, which in turn causes the primary system to cool down (see Figure 14.1.3-3). With a negative MTC (see Figure 14.1.3-2), the primary system cooldown causes the reactor power level to increase (see Figure 14.1.3-1. However, due to power decalibration, the indicated nuclear power level does not increase along with the reactor power level. Eventually, the indicated thermal power level reaches the Variable Overpower reactor trip ceiling, and the reactor is tripped. This terminates the power excursion. 98-182

Insert A

The minimum DNBR for the limiting Increase in Steam Flow case (with 51.8% excess steam flow) is calculated to be 1.36, which is well above the 95/95 DNBR safety limit. ~~Also, the peak linear heat rate (LHR) for that case is calculated to be 18.5 kW/ft, which is well below the fuel centerline melt criterion of 21.0 kW/ft.~~ These results demonstrate that fuel failures do not occur for the Increase in Steam Flow event and that the event acceptance criteria are satisfied.

The responses of key system variables are given in Figures 14.1.3-1 to 14.1.3-7. The sequence of events is given in Table 14.1.3-4. The MDNBR and the peak reactor power level calculated for each of the Increase in Steam Flow cases analyzed are listed in Table 14.1.3-5.

14.1.3.7 Conclusion

The results of the analysis demonstrate that the event acceptance criteria are met since the minimum DNBR predicted for the full power case is greater than the safety limit. The correlation limit assures that with 95% probability and 95% confidence, DNB is not expected to occur; therefore, no fuel is expected to fail. The ~~fuel centerline melt threshold of 21 kW/ft~~ is not violated during this event. (5/90)
FCM LHR limit

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Insert "A"

The maximum linear heat generation rate was not calculated for this event because is not a limiting moderate frequency event with respect to challenging the fuel centerline melt criterion of 21 kW/ft. Because the limiting moderate frequency events do not violate the fuel centerline melt criteria, it is concluded that this event will not violate the fuel centerline melt criterion.

This event does not challenge the FCMLHR limit. Therefore, LHR is not evaluated.

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14.1.5.1.6.1.1 Secondary System Parameters

Upon break initiation the break flow increased sharply and then began to decline in response to falling secondary side pressure. When the turbine trip occurred, the break flow increased due to a local pressure increase. The main steam line flow rate from each generator initially increased (see Figure 14.1.5.1-6) in response to the break and the assumed instantaneous full opening of the turbine control valves. The increased steam flow creates a mismatch between the core heat generation rate and the steam generator heat removal rate. This power mismatch causes the primary-to-secondary heat transfer rate to increase, which in turn causes the primary system to cool down (see Figure 14.1.5.1-2). When the reactor scram occurred, the turbine valves closed and steam flow declined sharply. At this point, the MFW flow may exceed the steam flow as the control system attempts to restore steam generator mass. Both steam flow and MFW flow were terminated when the main steam isolation valves closed.

14.1.5.1.6.1.2 Primary System Parameters

Approximately five seconds after the break occurred, the core inlet temperature began to decline. With a negative MTC (see Figure 14.1.5.1-3), the primary system cooldown caused the reactor power level to increase. The core power continues to increase until reactor scram on low steam generator pressure occurs. This terminated the power excursion. The pressurizer pressure and level began to decline as the volume of water in the primary system shrank. The core inlet mass flow rate increased due to the increasing density of the primary system fluid while the reactor coolant pumps' speed remained constant.

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14.1.5.1.6.1.3 Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results

The MDNBR value for this scenario was calculated to be 1.298 which is above the 95/95 XNB correlation limit. Therefore, no fuel rods would be expected to fail during this transient scenario from an MDNBR stand point.

The peak LHR for the LHR-limiting case (3.50 ft^2 break outside containment and downstream of a check valve) is calculated to be 18.7 kW/ft. Comparing this LHGR value with the FCMLHR limit ~~a centerline melt criteria of 21 kW/ft~~, it is apparent that centerline melt is not predicted to occur. Thus, no fuel failures are predicted to occur due to violation of the centerline melt criteria.

14.1.5.1.6.2 Hot Full Power 3.51 ft^2 Inside Containment Asymmetric Break Concurrent with a Loss of Offsite Power

The ANF-RELAP NSSS simulation of the most limiting pre-scram SLB scenario from an MDNBR standpoint (i.e., HFP 3.51 ft^2 inside containment asymmetric break concurrent with a loss of offsite power) is illustrated in Figures 14.1.5.1-7 through 14.1.5.1-11. A tabulation of the sequence of events is presented in Table 14.1.5.1-8. The ANF-RELAP computation was terminated 60 seconds after break initiation. This is well beyond the time of MDNBR or peak LHGR.

The transient is initiated by the opening of the break. The RCPs tripped shortly after transient initiation. The sharp reduction in the reactor coolant flow causes this pre-trip pumps off calculation to become a heat up transient very similar to a Loss of Coolant Flow event. Typically, the Steam Line Break calculation is a cooldown event. Because this case is a heat up event the most positive BOC neutronics conditions are used, and the maximum inside containment asymmetric break size is used. The maximum break size causes the biggest decrease in primary pressure. Maximizing the primary system pressure decrease causes the maximum decrease in moderator density and the maximum positive moderator feedback. The RCP trip causes the RCS flow to decrease rapidly throughout this transient. The decreasing RCS flow causes the transient time of the fluid in the core to increase and the fluid temperature begins to rise. The increasing fluid temperature causes positive moderator feedback, which in turn causes an increase in core power. However, the decreasing RCS flow causes the heat transfer to the fluid to decrease. The increase in core power is offset by the decrease in heat transfer from the fuel rods, such that, the fuel rod heat flux decreases slightly until reactor scram. The reactor scrams on the low reactor coolant flow trip signal.

14.1.5.1.6.2.3 Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results

The MDNBR value for the pre-scram 3.51 ft² asymmetric break inside containment with a loss of offsite power was calculated to be 0.88 which is below the 95/95 XNB correlation limit. The number of failed assemblies is determined by comparing the core power distribution to the assembly power where DNB occurs. This results in a predicted failure of 3.7% of the fuel rods in the core. 98-145

The peak LHR for this case is bounded by the 3.50 ft² outside containment symmetric break. Therefore, the LHGR for this case is below the ~~criteria of 21.0 kW/ft~~ and no fuel failures are predicted to occur due to violation of the centerline melt criteria.
FCM LHR limit

14.1.5.1.7 Conclusions

The HFP 3.50 ft² break outside containment and downstream of a check valve (symmetric break) with offsite power available was determined to be the most limiting in this analysis from an LHGR standpoint (19.7 kW/ft). In no scenario evaluated, however, was fuel failure calculated to occur as a result of violating the ~~21 kW/ft fuel centerline melt criteria.~~
FCM LHR limit

The HFP 3.51 ft² asymmetric break inside containment coincident with a loss of offsite power was determined to be the most limiting in this analysis from the standpoint of MDNBR. The MDNBR was calculated to be 0.88 which is below the 95/95 XNB correlation limit. This results in a predicted failure of 3.7% of the fuel rods in the core.

14.1.5.2 Post-Scram Analysis

14.1.5.2.1 Event Initiator

This event is initiated by a rupture in the main steam piping downstream of the integral steam generator flow restrictors and upstream of the MSIVs which results in an uncontrolled steam release from the secondary system.

PRISM

The analysis of the peak LHGR also comes from the ~~XTGPWR~~ PRISM and XCOBRA-IIIC analysis. The peak LHGR is calculated from the ANF-RELAP total core power and the ~~XTGPWR~~ PRISM radial and axial peaking. The peak LHGR, ~~24.27 kW/ft,~~ was calculated for the ~~HFP~~ HZP outside containment break with offsite power available event. ~~When compared to a centerline melt criteria of 21.0 kW/ft, four assembly quadrants (one full assembly) or 0.46% of the core, are predicted to fail due to violation of the centerline melt criteria.~~

Insert 6

14.1.5.2.6.2 Hot ~~Full~~ Zero Power Outside Containment with Loss of Offsite Power

The ANF-RELAP NSSS simulation of the most limiting SLB scenario from an MDNBR standpoint (i.e., ~~HFP~~ HZP outside containment break with a loss of offsite power) is illustrated in Figures 14.1.5.2-10 through 14.1.5.2-16. A tabulation of the sequence of events is presented in Table 14.1.5.2-8. Termination of the AFW by manual operator action was assumed to occur 600 seconds after initiation of the break. This is well beyond the time of MDNBR and maximum LHGR. ~~Termination of AFW would cause the affected SG to dry out and an increase in the primary system temperature. The increase in primary temperature, will drive the reactor subcritical and restore shutdown.~~

14.1.5.2.6.2.1 Secondary System Thermal Hydraulic Parameters

Steam flow out the break is the source of the NSSS cooldown. Steam flow for the affected steam generator is plotted in Figure 14.1.5.2-10. Secondary pressure for the steam generators is plotted in Figure 14.1.5.2-11. The affected steam generator blows down through the break throughout the transient. The pressure and mass flow rate dropped rapidly at first and then proceeded downward at a slower decay rate until natural circulation flow was established by approximately ~~250~~ 220 seconds. 98-14

The intact steam generators blow down for a short period until the MSIVs completely close approximately ~~16~~ 14 seconds after the break is initiated. The pressure recovers as the intact steam generator equilibrates with the primary system. Subsequently, the intact steam generator pressure remains essentially constant as the primary intact coolant loop approaches natural circulation conditions.

14.1.5.2.6.2.2 Primary System Thermal Hydraulic Parameters

The primary system core coolant temperature and pressure, pressurizer pressure and level responses resulting from the break flow are illustrated in Figures 14.1.5.2-12 through 14.1.5.2-14. The primary system pressure decays rapidly as the coolant contracts due to the cooldown and the pressurizer empties. Continued pressure reduction in the primary system causes the relatively hot stagnant liquid in the head of the RPV vessel to flash. The flashing in the upper head, coupled with near equilibration of other NSSS parameters, retards the pressure decay from that point forward.

A comparison of intact and affected core sector inlet temperatures throughout the transient indicates significant differences due to the limited cross flow allowed between loops. The core sector flows all show the same trend due to the coastdown of the primary coolant pumps. That is, all flows decrease rapidly until natural circulation conditions are achieved in the two flow loops.

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INSERT

6 ~~, 0.22 % of the core is predicted to fail. For~~
, less than ~~conservatism,~~ one full assembly, or 0.46% of the core ~~will be assumed~~ is predicted to fail due to the violation of the fuel centerline melt criteria when determining the radiological consequences of a main steam line break.

INSERT 6

No fuel failure is predicted to occur due to the violation of the FCMLHR limit. However, one full assembly, or 0.46% of the core, is assumed to fail when determining the radiological consequences of a main steam line break.

14.1.5.2.6.2.3 Reactivity and Core Power

The reactivity transient calculated by ANF-RELAP is illustrated in Figure 14.1.5.2-15. Initially, the core is assumed to be at full power. All control rods, except the most reactive one, are assumed to be inserted into the core following the reactor trip signal. The reactivity transient then proceeds. The total core reactivity, initially at 0.00¢, decreased instantly due to the scram worth at reactor trip, but then steadily increased due to moderator and Doppler feedback associated with the primary system cooldown. Shortly thereafter, power begins to rise steadily due to the dominating positive reactivity feedback from the moderator. The reactor soon achieves a quasi-steady-state power level where the Doppler and the moderator reactivities balance the scram reactivity.

Ninety seconds after break initiation, the RCS pressure dropped below the shutoff head of the HPSI system and HPSI flow to the RCS began. But, the elevated primary pressure limited the delivery of boron into the core due to the pressure versus flow characteristics of the HPSI system and unborated water never cleared the safety injection lines during the transient.

The transient experienced by the core power is illustrated in Figure 14.1.5.2-16. The reactor power declined to a decay heat level during the first 150 seconds of the transient. The maximum peak power level of 207 MW or 7.7% of rated power occurred at 488 seconds.

14.1.5.2.6.2.4 XTGPWR and XCOBRA-IIIC Results

The XTGPWR calculation is initially made on the basis of ANF-RELAP predicted core power, flow, pressure, and inlet temperatures. The XTGPWR calculations provide the radial and axial power distributions for use in the XCOBRA-IIIC code. Due to the high power peaking in the region of the stuck control rod, and the low core average natural circulation flow rates, large moderator density decreases are calculated in several assemblies in this region in the XTGPWR calculation. This is a major factor in the flattening of the axial and radial profiles, and the significant reduction in reactivity observed when XTGPWR is compared to ANF-RELAP. An XCOBRA-IIIC analysis is also conducted to define the flow and enthalpy distribution within the high power assembly. 98-145

A comparison of the overall change in reactivity between ANF-RELAP and XTGPWR shows that ANF-RELAP conservatively underestimates the negative reactivity by 1.00¢ at the time of MDNBR thus indicating that the ANF-RELAP power calculation is conservative.

14.1.5.2.6.2.5 Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results

The MDNBR of the hot fuel assembly is calculated to be 1.71 which is above the modified Barnett 95/95 DNBR correlation limit. Therefore, no fuel rods are expected to fail from an MDNBR standpoint.

As before, the analysis of the peak LHGR comes from the XTGPWR and the XCOBRA-IIIC analysis. The peak LHGR was 17.96 kW/ft. Comparing this LHGR with the ~~a centerline melt criteria of 21 kW/ft,~~ FCM LHGR limit it is apparent that centerline melt is not predicted to occur. Thus, no fuel failures are predicted to occur due to violation of the centerline melt criteria.

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14.1.5.2.7 Conclusions

The HFP and HZP scenarios, with offsite power maintained for operation of the primary coolant pumps resulted in a return to higher power levels than the scenarios where offsite power is lost. However, these scenarios provide substantially greater margin to the MDNBR limit because of the higher coolant flow rate. In no scenario evaluated, however, was fuel failure calculated to occur as a result of penetration of the MDNBR safety limit. The HFP and HZP scenarios with offsite power maintained for operation of the primary coolant pumps returned to higher power levels than the scenarios where offsite power is lost. Even though these scenarios have substantially greater margin to the MDNBR limit because of a higher coolant flow rate, the higher power levels in combination with the highly skewed power distribution due to the assumed stuck rod cluster resulted in them having the least margin to the fuel centerline melt limit.

The HFP outside containment break scenario concurrent with a loss of offsite power was determined to be the most limiting in this analysis from an MDNBR standpoint. The MDNBR of the hot fuel assembly is calculated to be 1.71 which is above the modified Barnett 95/95 DNBR correlation limit. Therefore, no fuel rods are expected to fail from an MDNBR standpoint.

The HFP outside containment break scenario with offsite power available was determined to be the most limiting in this analysis from the standpoint of centerline melt. This scenario results in the highest return to power and highest calculated LHGR of 24.27 kW/ft. ~~When compared to a centerline melt criteria of 21.0 kW/ft, four assembly quadrants (one full assembly) or 0.46% of the core, are predicted to fail due to violation of the centerline melt criteria.~~ which is below the FCM LHR limit. 48-145

14.1.5.3 Radiological Consequences of a Main Steam Line Break

The main steam line break is postulated to occur in a main steam line outside the containment. The radiological consequences of a main steam line break inside containment is bounded by the main steam line break outside containment. The plant is assumed to be operating with Technical Specification coolant concentrations and primary to secondary leakage. A 0.035 gpm primary to secondary leak is assumed to occur in both steam generators.

Two separate main steam line break cases are analyzed. In the first case, ~~associated with this accident is that~~ 1 fuel assembly is assumed to experiences melting and releases the melted fuel into the RCS at the onset of the accident. One fuel assembly is equivalent to 0.46% melt. The activity associated with the melt condition is therefore available for release to the atmosphere via primary to secondary leakage. In the second case a pre-accident iodine spike is assumed to occur. In this case the primary coolant iodine concentrations are 60 times the plant technical specification activity level of 1 uCi/gm DE I-131. In addition, the noble gas activity in the primary coolant is assumed to be at technical specification levels.

The noble gases and iodines in the primary coolant that leak into the faulted steam generator during the transient are released directly to the environment without holdup or decontamination. An iodine partition factor of 0.01 is used for the releases from the unaffected steam generator. Off-site power is assumed to be lost, thus making the condenser unavailable. The steam releases from the main steam line break are from the

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turbine building blowout panels as the atmospheric dispersion factor is greater for this release point than the enclosure building blowout panels. The steam releases from the intact steam generator are from the MSSVs/ADVs.

The radiological assuming one fuel assembly melted consequences of a main steam line break to the EAB, LPZ and Millstone 2 Control Room are reported in Tables 14.1.5.3-2 and 14.1.5.3-3. The assumptions used to perform this evaluation are summarized in Table 14.1.5.3-1. 98-145

The resulting doses to the EAB and LPZ do not exceed the limits specified in 10CFR100. The resulting doses to the Control Room do not exceed the limits specified in GDC19.

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TABLE 14.1.5.2-6

POST-SCRAM STEAM LINE BREAK ANALYSIS SUMMARY

Initial Power Level	Offsite Power Available	Break Location	Maximum Post-Scram Return to Power (MW)	MDNBR	Maximum LHGR (kW/ft)	Fuel Failure (% of Core)
HFP	No	outside containment	~~207.5~~ 109.3	~~1.71~~ 2.62	~~17.98~~ 10.3	0.0
HFP	Yes	outside containment	~~378.0~~ 194.8	~~2.28~~ 2.75	~~24.27~~ 21.0	0.0 ~~0.5~~ 0.005
HZP	No	outside containment	~~182.9~~ 132.1	~~1.89~~ 1.74	~~15.76~~ 16.76	0.0
HZP	Yes	outside containment	~~343.5~~ 271.6	~~2.37~~ 2.44	~~23.47~~ 23.3	0.0 ~~0.3~~ 0.22

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Initial Power Level	Offsite Power Availability	Break Location	ANF-RELAP Reactivity Change ($)	XTGPWR Reactivity Change ($)	Conservatisms in Input Parameters (MTC, Doppler, and Scram Worth Bias) ($)	Net Conservatism in ANF-RELAP model ($)
HFP	No	outside containment	+0.00	-6.30	+5.30	+1.00
HFP	Yes	outside containment	+0.00	-5.87	+4.86	+1.01
HZP	No	outside containment	+6.69	+3.00	+2.72	+0.97
HZP	Yes	outside containment	+6.68	+3.43	+2.34	+0.91

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TABLE 14.1.5.3-1

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ASSUMPTIONS USED IN MAIN STEAM LINE BREAK ANALYSIS

Core Power Level (MW_t)	2754
Primary to Secondary Leak Rate per Steam Generator	0.035 gpm
Primary Coolant Iodine Concentration	1 uCi/gm DE I-131
Secondary Coolant Iodine Concentration	0.1 uCi/gm DE I-131
Primary Coolant Noble Gas Concentration	$100/\bar{E}_{bar}$
Pre-accident Spike Iodine Concentration	60 uCi/gm DE I-131
Melted Fuel Percentage (assumed)	0.46%
Peaking Factor	1.45
Reactor Coolant Mass	430,000 lbs
Intact Steam Generator Minimum Mass	100,000 lbs
Safety Injection Signal Response	85 seconds
Site Boundary Breathing Rate (m^3/sec) 0 - 8 hr 8 - 24 hr 24 - 720 hr	 3.47E-04 1.75E-04 2.32E-04
Site Boundary Dispersion Factors (sec/m^3) EAB: 0 - 2 hr LPZ: 0 - 4 hr 4 - 8 hr 8 - 24 hr 24 - 96 hr 96 - 720 hr	 3.66E-04 4.80E-05 2.31E-05 1.60E-05 7.25E-06 2.32E-06
Control Room Breathing Rate	3.47E-04 m^3/sec
Control Room Damper Closure Time	5 seconds
Control Room Intake Prior to Isolation	800 cfm
Control Room Inleakage During Isolation	130 cfm
Control Room Emergency Filtered Recirculation Rate (t=10 min)	2,250 cfm
Control Room Intake Dispersion Factors (sec/m^3) PORVs/ADVs: 0 - 8 hr 8 - 24 hr 24 - 96 hr 96 - 720 hr	 3.19 E-03 2.05E-03 7.61E-04 2.13E-04
Turbine Building Blowout Panels: 0 - 8 hr 8 - 24 hr 24 - 96 hr 96 - 720 hr	 4.23E-03 2.85E-03 1.12E-03 3.63E-04
Control Room Free Volume	35,650 ft^3
Control Room Filter Efficiency (all iodines)	90%
Thyroid Dose Conversion Factors	ICRP 30

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TABLE 14.1.5.3-2

SUMMARY OF MILLSTONE 2 MSLB ACCIDENT DOSES
(Assuming 0.46% Melted Fuel)

Location	Thyroid (rem)	Whole Body (rem)	Beta (rem)
EAB	4.8	0.06	N/A
LPZ	2.3	0.02	N/A
Control Room	29	0.03	0.5

98-145

PORV setpoint. The peak ~~secondary~~ system pressure is 1086 psia. The resulting minimum DNBR is 1.59. Insert B

The responses of key system variables are given in Figures 14.2.1-1 to 14.2.1-5 for the primary system overpressurization case, Figures 14.2.1-6 to 14.2.1-11 for the secondary system overpressurization case, and Figures 14.2.1-12 to 14.2.1-17 for the minimum DNBR case. The sequence of events for each of these cases is given in Table 14.2.1-3, 14.2.1-4 and 14.2.1-5, respectively. 98-132 9/98

The primary and secondary side pressure relief valves have sufficient capacity to limit the respective system pressure to less than 110% (2750 and 1100 psia) of their design pressure.

14.2.1.7 Conclusion (5/90)

The calculated minimum DNBR for the Loss of Load event is above the heat flux correlation safety limit, so the Departure From Nucleate Boiling (DNB) SAFDL is not exceeded in this event. The peak pellet LHR is less than the ~~limit of 21 kW/ft~~ FCMLHR limit. The maximum primary and secondary system pressures remain below 110% of design pressure. Thus, the Loss of External Load event has been demonstrated to meet all required acceptance criteria. 98-132

14.2.2 Turbine Trip

14.2.2.1 Event Initiator

This event is initiated by a turbine trip which results in closure of the turbine stop valves and a rapid reduction in energy removal through the steam generators. 98-132

14.2.2.2 Event Description

The reactor protection system is designed to generate a reactor trip signal automatically when the turbine is tripped. Following reactor trip, there would be a rapid decrease in the energy being generated in the primary system. This would mitigate the consequences of the turbine trip event. Primary and secondary system overpressurization protection is provided by the code safety valves on both the primary and secondary systems and the secondary atmospheric dump valves. Also, if the condenser was available, the steam bypass system would be activated to reduce the secondary system pressure.

14.2.2.3 Reactor Protection

Reactor protection is provided by the high pressurizer pressure trip, variable overpower trip, TM/LP trip, low steam generator water level trip, and a nonsafety grade reactor trip on turbine trip. Additional protection is also provided by the primary and secondary side safety valves. Reactor protection for the Turbine Trip event is summarized in Table 14.2.2-1.

14.2.2.4 Disposition and Justification

This event is only credible for rated power and power operating conditions since the turbine will either be in tripped condition or there will be no load on the steam generators

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Insert to Page 14.2-3

Insert "B"

The maximum linear heat generation rate was not calculated for this event because it is not a limiting moderate frequency event with respect to challenging the fuel centerline melt criterion of 21 kW/ft. Because the limiting moderate frequency events do not violate the fuel centerline melt criteria, it is concluded that this event will not violate the fuel centerline melt criterion.

This event does not challenge the FCM LHR limit. Therefore, LHR is not evaluated.

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core was, however, redeveloped based on event specific XTGPWR (Reference 14.2-3) calculations. Core radial power distributions from full power EOC XTGPWR cases with differences between the hot-region inlet temperature and the cold-region inlet temperature are used to determine the power split between the halves of the core as a function of the difference in inlet temperatures. Since the temperature differences used in the XTGPWR cases meets or exceeds the inlet temperature difference calculated by ANF-RELAP during the transient calculation, the power splits used in ANF-RELAP are bounding. The XTGPWR calculations for the single MSIV closure event differ from those used in the SLB event. The SLB analysis requires power distribution data for all rods inserted minus the most reactive stuck rod, whereas in the single MSIV closure event it is assumed that an all rods out power distribution is appropriate. 98-132 9/98

The limiting results were obtained from the case with the lower steam flow rates. The results of the limiting EOC analysis are given in the event summary, Table 14.2.4-3, and in Figures 14.2.4-1 through 14.2.4-5. As indicated in the event summary table the secondary safety valves open early in the transient limiting the temperature rise on the hot side of the core associated with the closed MSIV. The reactor trips on low steam generator pressure which terminates the power rise. (5/90)

The peak LHR and Minimum Departure From Nucleate Boiling Ratio (MDNBR) are predicted to occur on the cold side of the core at the time of reactor trip. The peak LHR is 18.7 kW/ft and the deterministic MDNBR was found to be 1.40. Thus it is concluded that the DNB limits will not be violated and that fuel failures are precluded during the single MSIV closure event. 98-132 (93-18 4/93)

The secondary side safety valve setpoints were modeled with a ±3% drift allowance, and the flow characteristics were modeled with a 3% allowance for accumulation. The maximum secondary side pressure is 1092 psia, which is less than 110% (1100 psia) of design pressure. 98-132

14.2.4.7 Conclusion

The calculated minimum DNBR for the single MSIV closure event is above the critical heat flux correlation safety limit, so the DNB SAFDL is not exceeded in this event. The peak LHR is less than the ~~21 kW/ft~~ FcnLHR limit ~~to centerline melt~~. The maximum secondary side pressure is below 110% of design pressure. Thus, the single MSIV closure event has been demonstrated to meet all required acceptance criteria. 98-132

14.2.5 Steam Pressure Regulator Failure

Millstone Unit 2 does not have any steam line pressure regulators, so this event is not credible for this plant. No analysis needs to be considered for this event.

14.2.6 Loss of Nonemergency AC Power to the Station Auxiliaries 98-132

This event is not in the current licensing basis for Millstone Unit 2 and therefore is not analyzed.

14.2.7 Loss of Normal Feedwater Flow

The rated power case is bounding because of the reduced DNB margin for this initial state combined with the highest power to flow ratio during coastdown.

For the two pump loss of flow cases, the magnitude of the coastdown is less severe than the four pump coastdown, and the consequences of this event are bounded by the four pump loss of flow event. For the two pump flow coastdown cases, there is always some degree of forced reactor coolant flow. These events are, therefore, not as challenging as the four pump coastdown events. A comparison of the governing parameters indicates that these events are bounded by the four pump loss of flow event from full rated power conditions.

In summary, the four pump loss of flow event is the bounding event for the 14.3.1 events in all modes of operation. The only active system challenged is the reactor protection system (RPS) which is redundant and single failure proof.

The disposition of events for the Loss of Forced Reactor Coolant Flow event is summarized in Table 14.3.1-2.

14.3.1.5 Definition of Events Analyzed

This event is analyzed from full power initial conditions. The core thermal margins are minimized at full power conditions resulting in this being the bounding mode of operation for this event. One case is analyzed for this event to assess the challenge to the DNB Specified Acceptable Fuel Design Limit (SAFDL). 98-152 10/98

The loss of coolant flow immediately results in a loss of system heat rejection capacity. This causes the primary system coolant temperature to increase. The objective of selecting input and biasing is to minimize Departure From Nucleate Boiling Ratio (DNBR). The event analysis is, therefore, biased to minimize pressure which minimizes DNBR. The steam bypass and the atmospheric dump valves are both assumed not to operate, which again most challenges the DNB SAFDL. (93-18) (4/93)

14.3.1.6 Analysis Results

The transient is initiated by tripping all four primary coolant pumps. As the pumps coast down, the core flow is reduced, causing a reactor scram on low flow. No credit was taken for the RCP under-speed trip. As the flow coasts down, primary temperatures increase. This increase in temperature causes a subsequent power rise due to moderator reactivity feedback. The primary challenge to DNB is from the decreasing flow rate and resulting increase in coolant temperatures.

The DNBR consequences of this event were evaluated using Siemens Power Corporation (SPC) statistical setpoint methodology (Reference 14.3-1). The event Minimum Departure From Nucleate Boiling Ratio (MDNBR) was shown to be greater than thermal margin limits. The deterministic minimum DNBR is 0.95. ~~The peak pellet Linear Heat Rate (LHR) is calculated to be 19.5 kw/ft~~ INSERT "D" (93-18) 98-152 (93-18)

The responses of key system variables for the deterministic case are given in Figures 14.3.1-1 to 14.3.1-7. The sequence of events is given in Table 14.3.1-3. (5/91)

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Insert to Page 14.3-2

Insert "D"

The maximum linear heat generation rate was not calculated for this event because it is not a limiting moderate frequency event with respect to challenging the fuel centerline melt criterion of 21 kW/ft. Because the limiting moderate frequency events do not violate the fuel centerline melt criteria, it is concluded that this event will not violate the fuel centerline melt criterion.

This event does not challenge the FCMLHR limit. Therefore, LHR is not evaluated.

14.3.1.7 Conclusion

The statistical setpoint analysis demonstrates that the MDNBR limit is not penetrated by the Loss of Forced Reactor Coolant Flow event. Maximum peak pellet LHR for this event is below the ~~incipient fuel centerline melt criterion of 21 kw/ft.~~ FCM LHR limit

14.3.2 Flow Controller Malfunction

There are no flow control devices on the primary RCS of Millstone Unit 2. This event is therefore not credible and need not be analyzed.

14.3.3 Reactor Coolant Pump Rotor Seizure

14.3.3.1 Event Initiator

This event is initiated by an instantaneous seizure of an RCP rotor.

14.3.3.2 Event Description

The RCP seizure causes an immediate reduction in RCS flow rate. As in the Loss of Forced Coolant Flow event (Event 14.3.1), the impact of losing an RCS pump is a decrease in the active flow rate in the reactor core and an increase in core temperatures. Prior to reactor trip, the combination of decreased flow and increased temperature poses a challenge to DNB limits. A pressurization of the primary system will also occur due to the heatup of the primary coolant which causes a rapid insurge into the pressurizer.

14.3.3.3 Reactor Protection

Reactor protection for the RCP rotor seizure event is provided by the low reactor coolant flow trip, TM/LP trip, and the high pressurizer pressure trip.

Reactor protection for the Reactor Coolant Pump Rotor Seizure event is summarized in Table 14.3.3-1.

14.3.3.4 Disposition and Justification

This event is a concern for only rated power and power operating conditions because for other reactor operating conditions there is sufficient thermal margin so there will not be a challenge to the fuel design limits. The core heat flux to flow ratio is an excellent indicator of the potential DNB challenge for a loss of flow event. The highest ratios for this event are predicted to occur during the first few seconds of the transient from full power rated operating conditions. The consequences of this event will therefore be bounded by a pump rotor seizure event initiated from full power rated conditions. There is no single failure considered which could worsen the results.

The disposition of events for the Reactor Coolant Pump Rotor Seizure event is summarized in Table 14.3.3-2.

14.3.3.5 Definition of Events Analyzed

One case is analyzed for this event to maximize the challenge to the DNB limit. The bounding operating mode for this event is full power initial conditions. (93-18) (4/93)

14.3.3.6 Analysis Results

The locked rotor analysis assumes the locked pump loss coefficient given by the ~~homolo-~~ gous pump curves at zero pump speed. ~~The peak pellet LHR for this event is 19.6 kw/ft.~~ The sequence of events is given in Table 14.3.3-3 and the responses of key system variables are given in Figures 14.3.3-1 to 14.3.3-7 for the deterministic case. INSERT E 98-152 10/98 (98-18) (5/91) (93-18)

The DNBR consequences of the Loss of Flow event (14.3.1), which has a deterministic MDNBR of 0.95, were evaluated using SPC statistical setpoint methodology (Reference 14.3-1), and the MDNBR was shown to be greater than thermal design limits. Because the Rotor Seizure event (14.3.3) has a deterministic MDNBR of 1.01 and is inherently similar to the Loss of Flow event, penetration of thermal design limits is precluded for the Rotor Seizure event, as well. 98-152

14.3.3.7 Conclusion

The MDNBR limits are not exceeded by this event. The peak LHR is less than the ~~21 kw/ft~~ FCM LHR limit. ~~to centerline melt.~~ (5/91) 98-152

14.3.4 Reactor Coolant Pump Shaft Break

This event is not in the current licensing basis for Millstone Unit 2 and is, therefore, not analyzed.

Insert to Page 14.3-4

Insert "E"

~~The maximum linear heat generation rate was not calculated for this event because it is not a limiting event with respect to challenging the fuel centerline melt criterion of 21 kW/ft. Thus, it is concluded that this event will not violate the fuel centerline melt criterion.~~

This event does not challenge the FCMLHR Limit. Therefore, LHR is not evaluated.

BOC to end of cycle (EOC), for a spectrum of reactivity insertion rates. The only active system challenged by this event is the RPS, which is redundant and single failure proof.

The disposition of events for the Uncontrolled Control Rod/Bank Withdrawal at Power event is summarized in Table 14.4.2-2. (5/10)

14.4.2.5 Definition of Events Analyzed

The analysis evaluates the consequences of an uncontrolled control rod bank withdrawal from rated power. A spectrum of reactivity insertion rates were evaluated in order to bound events ranging from a slow dilution of the primary system boron concentration to the fastest allowed control bank withdrawals. Specifically, the analysis encompasses reactivity insertion rates from 4×10^{-6} to 4×10^{-4} delta rho/sec. 98-149 10/98

14.4.2.6 Analysis Results

The uncontrolled control bank withdrawal transients were analyzed for full power conditions (100% of rated). The limiting uncontrolled control rod bank withdrawal at 100% power occurred with EOC kinetics at an insertion rate of 4×10^{-6} delta rho/sec. The MDNBR was calculated as 1.21. This transient tripped on a TM/LP signal. The maximum peak pellet linear heat rate (LHR) occurs in a 100% power case which uses BOC kinetics. The maximum peak pellet LHR is calculated to be ~~20.0 kw/ft.~~ less than ~~21 kw/ft~~ the FCM LHR limit. 98-149 98-149

The sequence of events for the Uncontrolled Bank Withdrawal transient is given in Table 14.4.2-3. The transient response of key system variables are given in Figures 14.4.2-1 to 14.4.2-6.

14.4.2.7 Conclusion

Reactivity insertion transient calculations demonstrate that the XNB correlation limit will not be penetrated during any credible reactivity insertion transient at full power. The maximum peak pellet linear heat generation rate for this event is less than the ~~fuel centerline melt criterion of 21 kw/ft~~ FCM LHR limit. Applicable acceptance criteria are therefore met, and the adequate functioning of the TM/LP trip is demonstrated.

14.4.3 Control Rod Misoperation

The control rod misoperation event encompasses a number of transients resulting from different event initiators. The specific events addressed under this event category include the following:

(1) Dropped control rod or control rod bank;

(2) Dropped part-length control rod;

(3) Malpositioning of the part-length control rod group;

(4) Statically misaligned control rod/control rod bank;

The disposition of events for the Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve event is summarized in Table 14.6.1-2.

14.6.1.5 Definition of Events Analyzed

As discussed above, this event is analyzed for Minimum Departure from Nucleate Boiling Ratio (MDNBR) for both Modes 1 (full power), and 2 (startup). The startup power case is analyzed because the TM/LP trip can be manually bypassed below 5% power.

The system response for the full power case was evaluated by using PTSPWR2 (Reference 14.6-1). The full power event MDNBR was calculated using XCOBRA-IIIC (Reference 14.6-2).

The system response for the startup case was determined by conservative problem constraints. The maximum power was limited to 7% of the rated power. Above this power the assumed TM/LP trip bypass is automatically removed. The system pressure is conservatively assumed to be at the core inlet saturation pressure. The core inlet temperature is assumed to be at a level consistent with a maximum power rise of 7% and a conservative time delay before the SIS terminates the event. XCOBRA-IIIC was used with these system responses to predict the hot channel mass flux required for the critical heat flux calculation. The thermal margin was conservatively determined by the Modified Barnett critical heat flux correlation (Reference 14.6-3), with the system pressure reduced to the 725 psia upper limit of the Modified Barnett correlation. 98-200 98-200

14.6.1.6 Analysis Results

The sequence of events for the full power analysis are given in Table 14.6.1-3. Figures 14.6.1-1 to 14.6.1-6 show the transient response for key system variables. The MDNBR for this event initiated from ~~full power~~ is 1.20. ~~The peak Linear Heat Rate (LHR) is calculated to be 19.2 kw/ft.~~ INSERT "F" 98-200

The startup mode case resulted in a minimum critical heat flux ratio of above 10, as calculated by the Modified Barnett correlation. The peak pellet LHR is less than the full power value. Thus, the startup mode is bounded by the full power mode.

The charging and SISs have been shown to have sufficient capacity to easily compensate for the loss of primary coolant mass through the inadvertent opening of the pressurizer pressure relief valves. Therefore, the core is not expected to uncover during this event. 98-200

14.6.1.7 Conclusions

The results of the analysis demonstrate that the event acceptance criteria are met since the MDNBR predicted for the full power case is greater than the XNB correlation safety limit and the minimum Critical Heat Flux Ratio (CHFR) predicted for the startup mode case is greater than the Modified Barnett Critical Heat Flux (CHF) limit. The correlation limits assure with 95% probability and 95% confidence, that DNB is not expected to occur; therefore, no fuel is expected to fail. The ~~fuel centerline melt threshold of 21.0 kw/ft~~ FCMLHR limit is not violated in this event.

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Insert to Page 14.6-2

Insert "F"

The maximum linear heat generation rate was not calculated for this event because it is not a limiting moderate frequency event with respect to challenging the fuel centerline melt criterion of 21 kW/ft. Because the limiting moderate frequency events do not violate the fuel centerline melt criteria, it is concluded that this event will not violate the fuel centerline melt criterion.

This event does not challenge the FCMLHR limit. Therefore, LHR is not evaluated.

Docket No. 50-336
B18185

Attachment 4

Millstone Nuclear Power Station, Unit No. 2

License Amendment Request - Unreviewed Safety Question
Proposed Revision to Final Safety Analysis Report
Fuel Centerline Melt Linear Heat Rate Limit (PLAR 2-00-2)
Change to The Technical Specifications Bases

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

the fuel centerline melt linear heat rate limit

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate at or less than 21 kw/ft. Centerline fuel melting will not occur for this peak linear heat rate. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the XNB correlation. The XNB DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.17. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature with four Reactor Coolant Pumps operating for which the minimum DNBR is no less than 1.17. The limits in Figure 2.1-1 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperatures is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 111.6% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in Table 2.2-1. The area of safe operation is below and to the left of these lines.