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Constellation Energy
R.E. Ginna Nuclear Power Plant

April 29, 2005

Ms. Donna M. Skay
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Subject: License Amendment Request Regarding Main Feedwater Isolation Valves
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Ms. Skay:

In accordance with the provisions of 10 CFR 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC) is submitting a request for a license amendment to modify the Technical Specifications (TS) for the R.E. Ginna Nuclear Power Plant.

The proposed amendment would revise the TS to allow the use of the main feedwater isolation valves in lieu of the main feedwater pump discharge valves to provide isolation capability to the steam generators in the event of a steam line break. The main feedwater isolation valves would have shorter stroke time requirements than the currently utilized main feedwater pump discharge valves and their location in closer proximity to the steam generators would reduce the available volume of water and consequently the available mass and energy available for release to the containment. Additional conforming changes to the TS are included that take advantage of the improved plant configuration to establish greater consistency with the NRC approved Revision 3 to NUREG-1431, Standard Technical Specifications Westinghouse Plants.

It has been determined that this amendment application does not involve a significant hazard consideration as determined by 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

Enclosure 1 provides a description and assessment of the proposed changes. Enclosure 2 provides the existing TS pages marked up to show the proposed changes. Enclosure 3 provides

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revised (clean) TS pages. Enclosure 4 provides the existing TS Bases pages marked up to reflect the proposed changes (for information only). Changes to the TS Bases will be provided in a future update in accordance with the Bases Control Program. Enclosure 5 provides additional information regarding revised containment analysis performed supporting this amendment request. Enclosure 6 provides additional information regarding revised core response performed supporting this amendment request. Enclosure 7 provides a list of regulatory commitments associated with this license amendment request.

Approval of this amendment application is requested by April 30, 2006 to provide adequate time to prepare for implementation. Once approved, the amendment will be implemented prior to startup from the fall 2006 refueling outage.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated New York State Official.

If you have any questions regarding this submittal, please contact George Wrobel, Nuclear Safety and Licensing at (585) 771-3535.

Very truly yours,

Mary G. Korsnick

Mary G. Korsnick

Enclosures:
1. Evaluation of Proposed Change
2. Proposed Technical Specification Changes (markup)
3. Revised Technical Specification Pages (retyped)
4. Marked-up Copy of Technical Specification Bases
5. Revised Containment Analysis Results Using Gothic Computer Code
6. Steam Line Break at Hot Zero Power
7. List of Regulatory Commitments

STATE OF NEW YORK :
: TO WIT:
COUNTY OF WAYNE :

I, Mary G. Korsnick, being duly sworn, state that I am Vice President - R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC), and that I am duly authorized to execute and file this response on behalf of Ginna LLC. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Ginna LLC employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Mary G. Korsnick

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Monroe, this 29 day of April, 2005.

WITNESS my Hand and Notarial Seal:

Michalene A. Bunts
Notary Public

My Commission Expires:

1-11-2007
Date

MICHALENE A BUNTS
Notary Public, State of New York
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ENCLOSURE 1
R.E. Ginna Nuclear Power Plant

Description and Assessment of Proposed Change

Subject: Revision to Technical Specification 3.7.3 to Incorporate Replacement of Main Feedwater Pump Discharge Isolation Valves with Main Feedwater Isolation Valves

1.0 DESCRIPTION

This letter is a request to amend Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant.

This proposed change would revise the Operating License to change Technical Specification 3.7.3 to reflect a planned modification that is necessary to accommodate a planned power uprate for the facility. Approval of the amendment is requested by April 30, 2006 to provide adequate time to prepare for implementation. Once approved, the amendment will be implemented prior to startup from the fall 2006 refueling outage.

2.0 PROPOSED CHANGES

This request proposes to modify the Technical Specification (TS) by replacing the requirement in TS 3.7.3 that two Main Feedwater Pump Discharge valves (MFPDVs) be OPERABLE with the requirement that two Main Feedwater Isolation Valves (MFIVs) be OPERABLE. The time limit for closure of the MFIVs will be less than the current limit for the existing MFPDVs. The required Completion Times for closing or isolating Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulating Valves (MFRVs), and MFRV bypass valve(s) are being defined as 72 hours.

In addition, other changes to the specification, such as modifications to entry Conditions and Required Actions are being proposed that improve consistency with Revision 3 to NUREG-1431, Standard Technical Specifications Westinghouse Plants. These additional changes are based on the improved plant configuration resulting from the planned modification.

The proposed changes revise TS 3.7.3, Main Feedwater Regulating Valves (MFRVs), Associated Bypass Valves, and Main Feedwater Pump Discharge Valves (MFPDVs) as follows:

(a) The page headers for TS 3.7.3 currently state "MFRVs, Associated Bypass Valves, and MFPDVs."

The page headers for TS 3.7.3 are being revised to state "MFIVs, MFRVs, and Associated Bypass Valves."

(b) The title for TS 3.7.3 currently states "Main Feedwater Regulating Valves (MFRVs), Associated Bypass Valves, and Main Feedwater Pump Discharge Valves (MFPDVs)."

The title for TS 3.7.3 has been revised to state "Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulating Valves (MFRVs), and Associated Bypass Valves"

(c) TS 3.7.3 LCO currently states "Two MFRVs, associated bypass valves, and two MFPDVs shall be OPERABLE."

TS 3.7.3 LCO has been revised to state "Two MFIVs, two MFRVs, and associated bypass valves shall be OPERABLE."

(d) TS 3.7.3 Condition A currently states "One or more MFPDV(s) inoperable."

TS 3.7.3 Condition A is being revised to state "One or more MFIV(s) inoperable."

(e) TS 3.7.3 Required Action A.1 currently states "Close MFPDV(s)."

TS 3.7.3 Required Action A.1 is being revised to state "Close or isolate MFIV(s)."

(f) TS 3.7.3 Required Action A.2 currently states "Verify MFPDV(s) is closed."

TS 3.7.3 Required Action A.2 is being revised to state "Verify MFIV(s) is closed or isolated."

(g) The Completion Time for Required Actions A.1, B.1 and C.1 currently state "24 hours."

The Completion Time for Required Actions A.1, B.1 and C.1 is being revised to state "72 hours."

(h) Entry Condition D for TS 3.7.3 currently states "Required Action and associated Completion Time for Condition A, B, or C not met."

Entry Condition D for TS 3.7.3 is being revised to state "Required Action and associated Completion Time not met."

(i) Entry Condition E for TS 3.7.3 currently states

"One or more MFPDV(s) and one or more MFRV(s) inoperable.

<u>OR</u>

One or more MFPDV(s) and one or more MFRV bypass valve(s) inoperable."

Entry Condition E for TS 3.7.3 is being revised to state "Two valves in same flowpath inoperable."

(j) TS 3.7.3 Required Action E.1 currently states "Enter LCO 3.0.3."

TS 3.7.3 Required Action E.1 is being revised to state "Isolate affected flowpath."

(k) The Completion Time for Required Action E.1 currently states "Immediately."

The Completion Time for Required Action E.1 is being revised to state "8 hours."

(l) Entry Condition D and Required Actions D.1 and D.2 are being renumbered as E, E.1 and E.2 respectively.

(m) Entry Condition E and Required Action E.1 are being renumbered as D and D.1 respectively.

(n) The order of presentation of Actions D and E are swapped to reflect the renumbering of these items described above.

(o) TS 3.7.3 SR 3.7.3.1 currently states "Verify the closure time of each MFPDV is ≤ 80 seconds on an actual or simulated actuation signal."

TS 3.7.3 SR 3.7.3.1 is revised to state "Verify the closure time of each MFIV is ≤ 30 seconds on an actual or simulated actuation signal."

Changes to the TS Bases are provided in markup form as Enclosure 4. The changes to the Bases include necessary revisions that incorporate the proposed revised Specification wording, revise descriptive information to reflect the changes in associated plant equipment, remove information that is no longer appropriate, such as power supply information for the valves that are being eliminated from the Specification, and provide additional information relative to the planned plant modification that is consistent with information contained in the Bases for Revision 3 to NUREG-1431, Standard Technical Specifications Westinghouse Plants.

3.0 BACKGROUND

The valves specified in TS 3.7.3 close to isolate Main Feedwater (MFW) flow to the secondary side of the steam generators (SGs) following a Design Basis Accident (DBA). The safety related function of these valves is to provide for isolation of MFW flow to the secondary side of the SGs limiting the DBA for line breaks occurring downstream of the valves. Closure effectively terminates the addition of feedwater to an affected SG, limiting the mass and energy release for steam line breaks (SLBs) or feedwater line breaks (FWLBs) inside containment, and reducing the cooldown effects for SLBs.

The existing manually operated steam generator main feedwater inlet block valves (hereafter referred to as main feedwater isolation valves or MFIVs) located in the feedwater flowpaths to each steam generator will be utilized in lieu of the current main feedwater pump discharge valves to provide the isolation function. These valves will be modified by the addition of air actuators that will provide automatic closure capability upon receipt of an actuation signal. The actuation signal for the modified valves will be a safety injection signal. The closure time for these valves will be significantly less (30

seconds) than the closure time for the pump discharge valves (80 seconds). Since the MFIVs are located closer to the steam generators, the water volume downstream of the valves will be significantly reduced resulting in less mass and energy available for a shorter time for release inside containment.

The change to utilize the MFIVs in lieu of the feedwater pump discharge valves for feedwater isolation also results in an improved configuration in that the MFIVs are located in the individual flowpath to each steam generator. This configuration requires only the one MFIV in the flowpath to a steam generator to close to provide feedwater isolation in the event of a single failure of either the feedwater regulating valve and/or the associated bypass valve in that individual flowpath. The currently utilized feedwater pump discharge valves are located in parallel piping flowpaths upstream of a single common flowpath prior to the individual flowpaths to each steam generator. This configuration requires both pump discharge valves to close to provide feedwater isolation in the event of a single failure of either the feedwater regulating valve and/or the associated bypass valve in one flowpath to an individual steam generator.

4.0 TECHNICAL ANALYSIS

Closure of the MFIVs is necessary in the event of a failure of the main feedwater regulating valves to close on an SLB. The valve closure terminates the addition of feedwater to an affected SG, limiting the mass and energy release for SLBs or FWLBs inside containment, and reducing the cooldown effects for SLBs. The planned modification to utilize the MFIVs to perform this function provides a reduction in isolation valve closure time as well as reducing the water volume downstream of the isolation valves. Both of these changes significantly reduce the mass and energy release for SLBs or FWLBs inside containment as well as reduce cooldown effects for SLBs.

A SLB mass/energy release and containment response analysis has been performed in support of the planned power uprate for the R.E. Ginna plant and has determined changes are necessary to the existing plant design at the new power level. The containment response has been reanalyzed by Westinghouse using the GOTHIC computer code. Previous containment analysis for Ginna for SLBs has used the COCO computer code. Additional information regarding this new SLB mass/energy and containment analysis is provided as Enclosure 5. The new containment analysis also uses a different analytical value for the containment pressure hi-hi signal (initiates containment spray) than currently used. This change in the analysis incorporates additional margin for instrument uncertainty and will be addressed separately in the planned license amendment request for the extended power uprate. For purposes of this amendment, the use of the increased analytical value for containment hi-hi pressure signal is conservative since it delays the initiation of containment spray.

A core response analysis for steam system piping failures has also been performed in support of the planned power uprate. An analysis using the RETRAN computer code was performed in order to determine the plant transient conditions following a main

steam line break. The code models the core neutron kinetics, reactor coolant system (RCS), pressurizer, steam generators, safety injection (SI) system and the auxiliary feedwater (AFW) system; and computes pertinent variables, including the core heat flux, RCS temperature, and pressure. A conservative selection of those conditions were then used to develop core models which provide input to the detailed thermal and hydraulic digital computer code, VIPRE, to determine if the departure from nucleate boiling (DNB) design basis was met. The results of the analysis of a main steam pipe event indicate that the DNB design basis continues to be met. Therefore, this event does not adversely affect the core or the RCS, and applicable acceptance criteria are met. The analysis for core response at hot zero power (HZP) credits the operation of the MFIVs. Additional information regarding this new core response analysis for a steam line break at HZP is provided as Enclosure 6. The analysis at hot full power (HFP) is not included since it does not credit operation of the MFIVs.

Existing manual feedwater isolation valves, 3994 and 3995, are 14 inch ASME Safety Class 2 valves installed in the two 14 inch ASME Safety Class 2 pipelines which feed the steam generators. These manual valves will be modified to include installation of air actuated operators for both opening and closing. The source of air to each new actuator will be a dedicated accumulator for each of the valves. Each accumulator will be supplied from a non safety related instrument air header located in the vicinity of the valves. Each accumulator is being designed to accommodate a close-open-close cycle without makeup from the instrument air supply header.

The actuators will initiate valve closure on receipt of a safety injection (SI) signal. The SI signal to each actuator will be from a separate engineered safety feature actuation system (ESFAS) channel. The modified valves and actuator assemblies are being designed as Seismic Category 1. Each modified valve is being designed to open and be held open by its actuator and the actuators will be designed to fail closed on loss of instrument air supply or loss of electrical power. Each MFIV valve/actuator assembly is being designed to be suitable for closing under normal, upset and faulted conditions. The valve/actuator assemblies do not need to be qualified for a harsh environment since they are not required to function under such conditions (i.e., they are only credited with limiting design basis events that do not produce a harsh environment in the vicinity of the valves).

Each actuator will have an open/close switch and position indicating lights on the main control board. Control room alarms (via the plant process computer) are also being provided to alert operators of a low air pressure condition for either valve accumulator. Each actuator will be provided with a single train of safety related DC power for actuation, and non-safety related electrical connections for position indication and the low accumulator air pressure alarm. The DC power for each actuator will be from independent DC trains. Electrical separation criteria consistent with UFSAR section 8.3.1.4 are being utilized in the design of the modification.

The valves are being designed to close within 30 seconds of receipt of a safety injection signal which ensures FW isolation in less than the 32 seconds considered in the accident

analysis. The 32 seconds used in the accident analysis includes a 2 second delay for instrumentation response. This design also ensures the closing time is greater than 5 seconds, therefore minimizing the potential for water hammer affects.

The proposed TS changes described above include several conforming changes made possible by the change in plant configuration and are discussed below.

TS changes (a), (b), (c), and (d) are made to reflect use of generic terminology for the MFIVs. This change is consistent with Rev. 3 of NUREG 1431.

TS changes (e) and (f) are made to reflect the capability to otherwise isolate the MFIV(s) if valve closure is not possible. This provision is consistent with Rev. 3 of NUREG 1431 and affords greater flexibility in complying with the TS Required Action.

TS change (g) reflects an increased allowed Completion Time and is supported by the independence of each MFIV from the other steam generator's flowpath. The current parallel arrangement of the main feedwater pump discharge valves for this function requires both valves to be shut to compensate for the inoperability of any main feedwater regulating valve and its associated bypass valve. The new series arrangement requires only that the valve in series with the inoperable MFRV be closed. This change is consistent with Rev. 3 of NUREG 1431 that is based on this type of arrangement.

TS change (h) reflects the elimination (see TS change (j) of the Required Action (currently E.1) to enter LCO 3.0.3. The existing TS presentation for entry into current Condition D includes ". . . for Condition A, B, and C . . . ." This presentation exists solely to exclude the failure to meet current Required Action E.1 or its associated completion time as an entry condition for current Required Action D.1 or D.2. This change is made to reflect the elimination of the requirement to enter LCO 3.0.3 and is consistent with Rev. 3 of NUREG 1431.

TS change (i) reflects the change to plant configuration that eliminates the need for both MFIVs to be shut to compensate for the inoperability of any main feedwater regulating valve and its associated bypass valve. The proposed presentation provides greater simplicity while still providing an appropriate Condition Entry. This change is consistent with Rev. 3 of NUREG 1431.

TS changes (j) and (k) reflect NUREG 1431 recommended actions to isolate the affected flowpath in lieu of entry into LCO 3.0.3. The Bases of NUREG 1431 recognize that in this Condition there may be no redundant system to operate automatically and perform the required safety function and that the associated 8 hour Completion Time is reasonable to complete the actions required to close the valves, or otherwise isolate the affected flowpath. These changes are consistent with Rev. 3 of NUREG 1431.

Changes (l), (m), and (n) are based on the standard order of presentation based on Completion Times. These changes are consistent with Rev. 3 of NUREG 1431.

TS change (o) is made to reflect the reduction in closure time required for the MFIVs. This change is supported by the revised containment integrity analysis and the revised core response analysis and is consistent with Rev. 3 of NUREG 1431.

5.0 REGULATORY ANALYSIS

5.1 NO SIGNIFICANT HAZARDS CONSIDERATION

Ginna LLC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

Response: No

The proposed changes involve a modification to the plant configuration to ensure the acceptability of containment response for Steam Line Breaks (SLB) inside containment. The changes have also been evaluated to ensure the core response for steam system piping breaks remains acceptable. The changes to the Technical Specifications (TS) are necessary to properly accommodate the changes in plant configuration and ensure proper testing of the modified components.

The proposed changes do not adversely affect accident initiators or precursors nor significantly alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not adversely alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase the types and amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes cannot affect the probability of an accident occurring since they reflect a change in plant design consistent with current design which is not an accident initiator. The proposed changes cannot increase the consequences of postulated accidents since they reflect a change in plant design that will continue to mitigate the effects of feedwater addition to a faulted steam generator for a main steam line break inside containment. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

The proposed changes involve a modification to the plant configuration to ensure the acceptability of containment response for Steam Line Breaks (SLB) inside containment. The changes have also been evaluated to ensure the core response for steam system piping breaks remains acceptable. The changes to the Technical Specifications (TS) are necessary to properly accommodate the changes in plant configuration and ensure proper testing of the modified components.

The change in plant configuration significantly reduces the available water volume and therefore the mass and energy released to the containment in the event of an SLB with failure of a feedwater regulating valve. Existing feedwater flowpaths or piping are not significantly altered. An existing manual valve in the flowpath to each steam generator is utilized as the main feedwater isolation valve by the addition of an air actuator to provide automatic isolation capability. The changes do not involve a significant change in the methods governing normal plant operation. The TS changes modify the limiting condition for operation, required action statements, associated completion times and surveillance requirements to those that are consistent with those previously approved for Westinghouse plants in the Standard Technical Specifications found in NUREG-1431. The proposed TS changes do not create the possibility of a new or different type of accident from those previously evaluated since they reflect a design change that will accomplish the same feedwater isolation function as previously performed by the main feedwater pump discharge isolation valves with no significant change to the manner in which the feedwater system operates.

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

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

Response: No

The proposed changes involve a modification to the plant configuration to ensure the acceptability of containment response for Steam Line Breaks (SLB) inside containment. The changes have also been evaluated to ensure the core response for steam system piping breaks remains acceptable. The changes to the Technical Specifications (TS) are necessary to properly accommodate the changes in plant configuration and ensure proper testing of the modified components.

The level of safety of facility operation is unaffected by the proposed changes since there is no change in the intent of the TS requirements of assuring proper

main feedwater isolation in the event of a steam line break inside containment. The response of the plant systems to accidents and transients reported in the Updated Final Safety Analysis Report (UFSAR) is not adversely affected by this change. Therefore, the capability to satisfy accident analysis acceptance criteria is not adversely affected. The TS changes modify the limiting condition for operation, required action statements, associated completion times and surveillance requirements to those that are consistent with those previously approved for Westinghouse plants in the Standard Technical Specifications found in NUREG-1431. The proposed TS changes does not involve a significant reduction in the margin of safety since they are based upon a modification that will maintain the margin of safety with respect to feedwater addition for a main steam line break inside containment to the previously analyzed condition. Therefore, the changes do not involve a significant reduction in a margin of safety.

Based on the above, Ginna LLC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The regulatory requirements/criteria applicable to this request relative to containment integrity for this amendment are provided in Ginna UFSAR section 3.1.2.5.1, General Design Criterion 50 - Containment Design Basis. The criterion states, "The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters (GDC 50).

The reactor containment structure, penetrations, valves, access openings, and the containment spray system are designed with margin to accommodate the temperature and pressure conditions associated with the loss-of-coolant accident and main steam line break, without loss of function.

The design of the containment is based on a postulated main steam line break, feedwater line break, or a double ended rupture of a reactor coolant pipe, coupled

with partial loss of the redundant engineered safety features systems (minimum engineered safety features)."

The applicable portions of these criteria relative to this proposed license amendment is the containment response to a steam line break or a feedwater line break inside containment. The containment has been reanalyzed at the planned uprate power level utilizing the MFIVs in lieu of the MFPDVs providing feedwater isolation using the design valve closure time of 30 seconds. The analysis assumes the valves close in 32 seconds which includes 2 seconds for instrumentation response. The containment response in this analysis indicates that containment pressure remains below the containment design pressure and therefore maintains the margin of safety to the containment design pressure and associated environmental qualification parameters. The proposed TS specifies acceptance criteria for the closure times of these valves that will assure the assumptions in the MSLB inside containment analysis are met.

The core response has been reanalyzed at the planned uprate power level utilizing the MFIVs in lieu of the MFPDVs providing feedwater isolation using the design valve closure time of 30 seconds. The analysis assumes the valves close in 32 seconds which includes 2 seconds for instrumentation response. The acceptance criteria applicable to this request relative to core response are (1) GDC 27, insofar as it requires that the reactivity control systems are designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to ensure the capability to cool the core is maintained; (2) GDC 28, insofar as it requires that the reactivity control systems are designed to ensure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; (3) GDC 31, insofar as it requires that the reactor coolant pressure boundary is designed with sufficient margin to ensure that, under specified conditions, it will behave in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; and (4) GDC 35, insofar as it requires the reactor coolant system (RCS) and associated auxiliaries are designed to provide abundant emergency core cooling.

The steam release resulting from a main steam pipe rupture will result in an increase in steam flow, a reduction of coolant temperature and pressure, and an increase in core reactivity. The core reactivity increase can cause a power level increase and a decrease in shutdown margin.

Steam line breaks initiated from hot zero power (HZP) conditions have been conservatively analyzed at the planned uprated power level utilizing the MFIVs in lieu of the MFPDVs providing feedwater isolation using the design valve stroke time of 30 seconds. Steam line breaks initiated from hot full power (HFP) do not

credit operation of the MFIVs. The HFP transient is terminated via a reactor trip. The specific acceptance criteria applied for this event is (1) the departure from nucleate boiling ratio (DNBR) should remain above the 95/95 DNBR limit at all times during the transient. Demonstrating that the DNBR limit is met very conservatively demonstrates that GDC-27 requirements are met; (2) primary and secondary pressures must remain below 110% of their respective design pressures at all times during the transient. Demonstrating that the primary and secondary pressure limits are met including allowance made for the worst stuck rod satisfies the requirements of GDC-28 and GDC 31; and (3) The HZP case assumes emergency core cooling system (ECCS) actuation (i.e., safety injection (SI) flow) for mitigation. The analysis performed demonstrates that the SI system has sufficient capacity to mitigate the event. The analysis demonstrates that the requirements of GDC-35 are met.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in accordance with Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

Ginna LLC has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration; and

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluent that may be released offsite since there is no change in the type or quantities of material available for release than that previously analyzed; and

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure since the change in plant configuration does not significantly increase overall operations and maintenance requirements nor is any different type of equipment required to be installed.

Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.

7.0 REFERENCES

None

ENCLOSURE 2
R.E. Ginna Nuclear Power Plant

Proposed Technical Specification Changes (markup)

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Regulating Valves (MFRVs) and Associated Bypass Valves, ~~and~~ Main Feedwater ~~Pump Discharge~~ Isolation Valves ~~(MFPDVs)~~

LCO 3.7.3 Two MFRVs, associated bypass valves, and ~~two MFPDVs~~ MFIVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3 except when both steam generators are isolated from both main feedwater pumps.

ACTIONS

- NOTE -
Separate Condition entry is allowed for each valve. MFIVs

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ~~MFPDV(s)~~ inoperable.	A.1 Close ~~MFPDV(s)~~ or isolate	24 (72) hours
	AND	
	A.2 Verify ~~MFPDV(s)~~ is closed or isolated.	Once per 7 days
B. One or more MFRV(s) inoperable.	B.1 Close or isolate MFRV(s).	24 (72) hours
	AND	
	B.2 Verify MFRV(s) is closed or isolated.	Once per 7 days
C. One or more MFRV bypass valve(s) inoperable.	C.1 Close or isolate MFRV bypass valve(s).	24 (72) hours
	AND	
	C.2 Verify MFRV bypass valve(s) is closed or isolated.	Once per 7 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
D E Required Action and associated Completion Time ~~for Condition A, B, or C~~ not met.	~~D~~ E .1 Be in MODE 3. AND E ~~D~~ .2 Be in MODE 4.	6 hours 12 hours
E D ~~One or more MFPDV(s) and one or more MFRV(s)~~ inoperable. ~~OR~~ ~~One or more MFPDV(s) and one or more MFRV bypass valve(s) inoperable.~~	~~E~~ .1 ~~Enter LCO 3.0.3.~~ Isolate affected flowpath D	~~Immediately~~ 8 hours

swap order

Two valves in same flowpath

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the closure time of each ~~MFPDV~~ MFIV is ≤ ~~80~~ 30 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program
SR 3.7.3.2	Verify the closure time of each MFRV and associated bypass valve is ≤ 10 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program

ENCLOSURE 3
R.E. Ginna Nuclear Power Plant

Revised Technical Specification Pages (retyped)

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulating Valves (MFRVs), and Associated Bypass Valves.

LCO 3.7.3 Two MFIVs, two MFRVs, and associated bypass valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3 except when both steam generators are isolated from both main feedwater pumps.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MFIV(s) inoperable.	A.1 Close or isolate MFIV(s). AND A.2 Verify MFIV(s) is closed or isolated.	72 hours Once per 7 days
B. One or more MFRV(s) inoperable.	B.1 Close or isolate MFRV(s). AND B.2 Verify MFRV(s) is closed or isolated.	72 hours Once per 7 days
C. One or more MFRV bypass valve(s) inoperable.	C.1 Close or isolate MFRV bypass valve(s). AND C.2 Verify MFRV bypass valve(s) is closed or isolated.	72 hours Once per 7 days
D. Two valves in same flowpath inoperable.	D.1 Isolate affected flowpath	8 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	6 hours
	AND	
	E.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the closure time of each MFIV is ≤ 30 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program
SR 3.7.3.2	Verify the closure time of each MFRV and associated bypass valve is ≤ 10 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program

ENCLOSURE 4
R.E. Ginna Nuclear Power Plant

Marked-up Copy of Technical Specification Bases

the MFIVs are closed,

This Function is actuated by either a SG Water Level-High or an SI signal. The Function provides feedwater isolation by closing the Main Feedwater Regulating Valves (MFRVs) and the associated bypass valves. In addition, on an SI signal, the AFW System is automatically started and the MFW pump breakers are opened which closes the MFW pump discharge valves. The SI signal was discussed previously.

a. Feedwater Isolation-Automatic Actuation Logic and Actuation Relays

Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

Automatic initiation must be OPERABLE in MODES 1, 2, and 3. The Feedwater Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MFRVs and associated bypass valves are closed and de-activated or isolated by a closed manual valve. In MODES 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

b. Feedwater Isolation-Steam Generator Water Level-High

The Steam Generator Water Level-High Function must be OPERABLE in MODES 1, 2, and 3. The Feedwater Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MFRVs and associated bypass valves are closed and de-activated or isolated by a closed manual valve. In MODES 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

This signal provides protection against excessive feedwater flow. The ESFAS SG water level instruments have dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements. LT-461, LT-462, and LT-463 are the three channels required for SG A. LT-471, LT-472, and LT-473 are the three channels required for SG B. Each SG is considered a separate Function for the purpose of this LCO. The Allowable Value for SG Water Level-High is a percent of narrow range instrument span. The trip Setpoint is similarly calculated.

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Regulating Valves (MFRVs), Associated Bypass Valves and Main Feedwater Pump Discharge Valves (MFPDVs)

Isolation

BASES

(MFIVs)

(3994 and 3995)

BACKGROUND

The MFRVs (4269 and 4270) and their associated bypass valves (4271 and 4272), and MFPDVs (3977 and 3976) isolate main feedwater (MFW) flow to the secondary side of the steam generators (SGs) following a Design Basis Accident (DBA). The safety related function of the MFRVs, associated bypass valves, and MFPDVs is to provide for isolation of MFW flow to the secondary side of the SGs terminating the DBA for line breaks occurring downstream of the valves. Closure effectively terminates the addition of feedwater to an affected SG, limiting the mass and energy release for steam line breaks (SLBs) or feedwater line breaks (FWLBs) inside containment, and reducing the cooldown effects for SLBs.

MFIVs

in the flowpath

The MFRVs, associated bypass valves, and MFPDVs in conjunction with check valves located downstream of the isolation valves also provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact SG (see Figure B 3.7.3-1).

each MFW line to its respective SG, outside containment in the Intermediate Building.

MFIV

One MFPDV is located in the Turbine Building on the discharge line of each MFW pump (Ref. 1). One MFRV and associated bypass valve is located on each MFW line to its respective SG, outside containment in the Turbine Building. The MFRVs, associated bypass valves, and MFPDVs are located upstream of the AFW injection point so that AFW may be supplied to the SGs following closure of the MFRVs and bypass valves. The piping volume from these valves to the SGs is accounted for in calculating mass and energy releases, and must be refilled prior to AFW reaching the SG following either an SLB or FWLB.

MFIVs

and MFIVs

MFIVs

The MFPDVs closes on the opening of the MFW pump breaker which occurs on receipt of a safety injection signal or any other signal which trips the pump breaker. The MFRVs and bypass valves close on receipt of a safety injection signal, a SG high level signal, or on a reactor trip with T_{avg} < 554°F with the associated MFRV in auto. All valves may also be actuated manually. In addition to the MFRVs, associated bypass valves and MFPDVs, a check valve located outside containment for each feedwater line is available. The check valve isolates the feedwater line penetrating containment providing a containment isolation boundary.

MFIVs

air

The MFPDVs are motor operated valves which are supplied by MCC A and MCC B. These MCCs are supplied by Buses 13 and 15 respectively, which are in turn supplied by 4160 V Buses 11A and 11B. During normal plant operation, Buses 11A and 11B are supplied by Unit Auxiliary

~~Transformer 11 which is provided power directly from the Ginna Station turbine/generator. Following a turbine/generator trip and an approximately 60 second time delay, the feeder breakers to Buses 11A and 11B are tripped and bus tie breakers to Buses 12A and 12B, respectively, are closed. These bus tie breakers (52/BTA-A and 52/BTB-B) then supply Buses 11A and 11B from offsite power sources. The tie breakers are required for the MFPDVs to complete their closure as the valves have an approximate 72 second stroke time.~~

MFIVs

APPLICABLE SAFETY ANALYSES

The design basis of the MFRVs, associated bypass valves, and ~~MFPDVs~~ is established by the analyses for the SLB. The SLB is evaluated for two cases, one with respect to reactor core response and the second with respect to containment integrity (Ref. 2). The SLB for reactor core response is evaluated assuming initial conditions and single failures which have the highest potential for power peaking or departure from nucleate boiling (DNB). The most limiting single failure for this evaluation is the loss of a safety injection pump which reduces the rate of boron injection into the Reactor Coolant System (RCS) delaying the return to subcriticality. The MFRV and bypass valve on the intact SG for this case are assumed to close on a safety injection signal to prevent excessive cooldown of the RCS which could result in a lower DNB ratio. The failure of either of these valves is bounded by the eventual coastdown of the MFW pumps, which have their breakers opened by a SI signal, and the MFIVs ~~MFPDV~~ which close on ~~opening of the MFW pump breakers.~~ a safety Injection signal

The SLB for containment integrity is evaluated assuming initial conditions and single failures which result in the addition of the largest amount of mass and energy into containment. For this scenario, offsite power is assumed to be available and reactor power is below 100% RTP. With offsite power available, the reactor coolant pumps continue to circulate coolant, maximizing the RCS cooldown. At lower power levels, the SG inventory and temperature are at their greatest, which maximizes the analyzed mass and energy release to containment. The MFRV and bypass valve on the faulted SG are assumed to close on a safety injection signal to prevent continued contribution to the energy and mass released inside containment by the SLB. The failure of either of these valves is bounded by the eventual coastdown of the MFW pumps and closure of the ~~MFPDVs~~ MFIVs.

The MFRVs and bypass valves are also credited for isolation in the feedwater transient analyses (e.g., increase in feedwater flow). These valves close on either a safety injection or high SG level signal depending on the scenario. The valves also must close on a FWLB to limit the amount of additional mass and energy delivered to the SGs and containment.

The failure of the MFRVs to control flow is also considered as an initiating event. This includes consideration of a valve failure coincident with an atmospheric relief valve failure since a single component in the Advanced Digital Feedwater Control System (ADFCS) controls both components (Ref. 3). This combined valve failure accident scenario is evaluated with respect to DNB since a large RCS cooldown is possible with this combination of failures. However, this scenario is bounded by the SLB accident.

The MFRVs, associated bypass valves, and ~~MFPDVs~~ MFIVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO ensures that the MFRVs, associated bypass valves, and ~~MFPDVs~~ MFIVs will isolate MFW flow to the SGs, following a FWLB or SLB.

This LCO requires that two ~~MFPDVs~~ MFIVs, two MFRVs, and two MFRV bypass valves be OPERABLE. The MFRVs, associated bypass valves, and ~~MFPDVs~~ MFIVs are considered OPERABLE when isolation times are within limits and they can close on an isolation actuation signal. The isolation signal from reactor trip with T_{avg} < 554°F with the associated MFRV in auto is not a requirement for OPERABILITY.

~~In addition to the normal power supply, also included in the OPERABILITY of the MFPDVs are the 4160 V Bus 11A and 11B bus tie breakers to Bus 12A and 12B. These bus tie breakers (52/BTA-A and 52/BTB-B) provide power for closure of the MFPDVs following a turbine/generator trip.~~

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. It may also result in the introduction of water into the main steam lines for an excess feedwater flow event.

APPLICABILITY

The MFRVs, associated bypass valves, and ~~MFPDVs~~ MFIVs valves must be OPERABLE whenever there is significant mass and energy in the RCS and SGs. This ensures that, in the event of a DBA, the accident analysis assumptions are maintained. In MODES 1, 2, and 3, the MFRVs, associated bypass valves, and ~~MFPDVs~~ MFIVs are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated or isolated by a closed manual valve such that both SGs are isolated from both MFW pumps, they are already performing their safety function and no longer required to be OPERABLE.

In MODE 4, the MFIVs, MFRVs, and associated bypass valves, ~~and MFPDVs~~ are normally closed since AFW is providing decay heat removal due to the low SG energy level. In MODE 5 or 6, the SGs do not contain much energy because their temperature is below the boiling point of water; therefore, the MFIVs, MFRVs, associated bypass valves, and ~~MFPDVs~~ are not required for isolation of potential pipe breaks in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1

With one or more MFIV(s) ~~MFPDV(s)~~ inoperable, action must be taken to restore the affected valve to OPERABLE status, or close to or isolate the inoperable valve within ~~24~~ 72 hours. ~~The 24 hour Completion Time takes into account the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 24 hour Completion Time is reasonable, based on operating experience.~~

An inoperable MFIV ~~MFPDV~~ that is closed or isolated must be verified on a periodic basis that it remains closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion time is reasonable, based on engineering judgement, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

the redundancy afforded by the remaining OPERABLE valves and

B.1 and B.2

With one or more MFRV(s) inoperable, action must be taken to restore the affected valve to OPERABLE status, or to close or isolate the inoperable valve within ~~24~~ 72 hours. ~~The 24 hour Completion Time takes into account the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 24 hour Completion Time is reasonable, based on operating experience.~~

An inoperable MFRV that is closed or isolated must be verified on a periodic basis that it remains closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion time is reasonable, based on engineering judgement, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

C.1 and C.2

With one or more MFRV bypass valve(s) inoperable, action must be taken to restore the affected valve to OPERABLE status, or to close or isolate the inoperable valve within ~~24~~ 72 hours. The ~~24~~ hour Completion

Insert B 3.7.3-5A

With two inoperable valves in the same flow path, there may be no redundant system to operate automatically and perform the required safety function. Although the containment can be isolated with the failure of two valves in parallel in the same flow path, the double failure can be an indication of a common mode failure in the valves of this flow path, and as such, is treated the same as a loss of the isolation capability of this flow path. Under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the MFIV or MFRV, or otherwise isolate the affected flow path.

the redundancy afforded by the remaining OPERABLE valves and

Time takes into account the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 ~~24~~ hour Completion Time is reasonable, based on operating experience.

An inoperable MFRV bypass valve that is closed or isolated must be verified on a periodic basis that it remains closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

E ~~D.1 and D.2~~

If a Required Action and associated Completion time is not met,

~~If the MFRV, associated bypass valve, or MFPDV cannot be restored to OPERABLE status or closed within 24 hours or cannot be verified closed once per 31 days,~~ the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 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.

Swap order

F ~~E.1~~

Insert B 3.7.3-5A

~~If one or more MFPDV(s) and one or more MFRV(s), or one or more MFPDV(s) and one or more MFRV bypass valve(s) are inoperable, the plant is in a condition outside of the accident analyses; therefore, LCO 3.0.3 must be entered immediately. This Condition must be entered when any combination of MFRVs, associated bypass valves, or MFPDVs are inoperable such that a MFW pump, condensate pump, or condensate booster pump can provide unisolable flow to one or both SGs (see Figure B 3.7.3-1).~~

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV ~~MFPDV~~ is ≤ 30 ~~60~~ seconds from the full open position on an actual or simulated actuation signal (i.e., ~~from opening of MFW pump breakers~~). The valve closure times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. These valves should not be tested at power since even a partial stroke exercise increases the risk of a valve closure with the plant generating power. As these valves are not tested at power, they are exempt from the ASME Code, Section XI, (Ref. 4) requirements during operation in MODES 1, 2, and 3.

The Frequency for this SR is in accordance with the Inservice Testing Program.

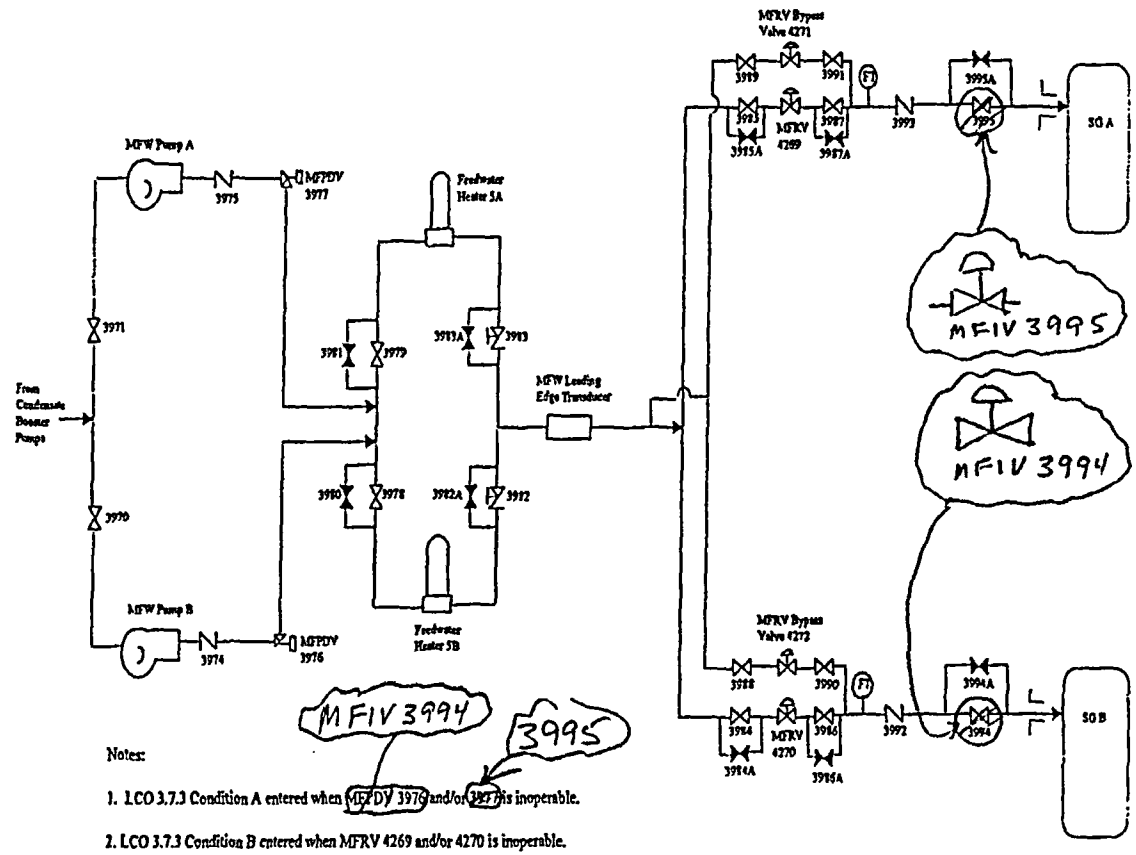
SR 3.7.3.2

This SR verifies that the closure time of each MFRV and associated bypass valve is ≤ 10 seconds from the full open position on an actual or simulated actuation signal. The valve closure times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. These valves should not be tested at power since even a partial stroke exercise increases the risk of a valve closure with the plant generating power. As these valves are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 4), requirements during operation in MODES 1, 2, and 3.

The Frequency for this SR is in accordance with the Inservice Testing Program.

REFERENCES

1. UFSAR, Section 10.4.5.3.
2. UFSAR, Section 15.1.5.
3. UFSAR, Section 15.1.6.
4. ASME, Boiler and Pressure Vessel Code, Section XI.



Notes:

1. LCO 3.7.3 Condition A entered when MFPDV 3976 and/or 3977 is inoperable.
2. LCO 3.7.3 Condition B entered when MFRV 4269 and/or 4270 is inoperable.
3. LCO 3.7.3 Condition C entered when MFRV Bypass Valve 4271 and/or 4272 is inoperable.
4. LCO 3.7.3 Condition D entered when any combination of valve inoperabilities results in an unusable flowpath from the condensate booster pumps to one or more SGs.

MFIV 3994 3995

two valves in same flowpath are inoperable.

Figure B 3.7.3-1
MFIVs, and MFRVs, Associated Bypass Valves ~~and MFPDVs~~

ENCLOSURE 5
R.E. Ginna Nuclear Power Plant

Steam Line Break Mass/Energy Release and Containment Response Analysis

Steam Line Break Mass/Energy Release and Containment Response Analysis

1.0 Introduction

Steamline ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment that could produce high pressure conditions for extended periods of time. The magnitude of the releases following a steamline rupture is dependent upon the plant initial operating conditions and the size of the rupture as well as the configuration of the plant steam system and the containment design. There are competing effects and credible single failures in the postulated accident scenario used to determine the worst cases for containment pressure following a steamline break.

The Ginna steamline break and containment response analysis considers a spectrum of cases that vary the initial power condition and the postulated single failure. The following sections identify the analysis methodology, the selection of cases, the major plant assumptions and the results of the analysis. Major elements considered in this analysis compared to the current licensing-basis analysis are the automatically-actuated main feedwater isolation valves (MFIVs) as a back-up to the feedwater regulating valves (MFRVs), the extended power uprate (EPU), and a reduction in the minimum required shutdown margin for the core at end of cycle conditions.

2.0 Analysis Methods and Computer Codes

This section identifies the methods and computer codes used to calculate the steamline break mass and energy releases and the containment pressure response.

2.1 Mass and Energy Release Methodology and Computer Code

The analysis documented herein uses the RETRAN code, which is documented in WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses" (Ref. 5).

The following limitations in the NRC SER for WCAP-14882-P-A have been adhered to in the use of RETRAN to analyze this event.

- The break flow model is the Moody model.
- Only steam (dry vapor) will exit the break, since perfect steam separation in the steam generators is assumed.
- Westinghouse will not evaluate the superheat in the steam released to the containment. Any superheated conditions will be reset to be equal to the saturation temperature.

2.2 Containment Response Methodology and Computer Code

The containment response analysis uses the GOTHIC computer code. The GOTHIC program is rapidly becoming the industry standard for performing containment pressure and temperature design-basis analyses. The GOTHIC Technical Manual (Ref. 6) provides a description of the governing equations, constitutive models, and solution methods in the solver. The GOTHIC Qualifications Report (Ref. 7) provides a comparison of the solver results with both analytical solutions and experimental data.

The Ginna GOTHIC containment evaluation model consists of a single lumped-parameter node; the diffusion layer model (DLM) is used for heat transfer to all structures in the containment. Plant input assumptions (identified in Section 4.4) are the same as, or slightly more restrictive, than in the current licensing-basis analysis performed with the COCO code (Ref. 8). Benchmarking between the Ginna COCO and GOTHIC models was performed to confirm consistency in the implementation of the plant input values.

This steamline break containment response analysis uses GOTHIC version 7.2. The latest code version is used to take advantage of the DLM heat transfer option. This heat transfer option was approved by the NRC (Ref. 9) for use in Kewaunee containment analyses with the condition that mist be excluded from what was earlier termed as the mist diffusion layer model (MDLM). The GOTHIC containment modeling for Ginna has followed the conditions of acceptance placed on Kewaunee. Kewaunee and Ginna both have large, dry containments with similar containment volumes and active heat removal capabilities. Changes in the GOTHIC code versions are detailed in Appendix A of the GOTHIC User Manual Release Notes (Ref. 10). Version 7.2 is used consistent with the restrictions identified in Ref. 9; none of the user-controlled enhancements added to version 7.2 were implemented in the Ginna containment model.

3.0 Case Definitions and Single Failures

There are many factors that influence the quantity and rate of the mass and energy release from the steamline. To encompass these factors, a spectrum of cases varies the initial power level and the single failure. This section summarizes the basis of the cases that have been defined for the Ginna plant.

The power level at which the plant is operating when the steamline break is postulated can cause different competing effects that make it difficult to pre-determine a single limiting case. For example, at higher power levels there is less initial water/steam in the steam generator, which is a benefit. However, at a higher power level there is a higher initial feedwater flowrate, higher feedwater temperature, higher decay heat, and there is a higher rate of heat transfer from the primary side, which are all penalties. Therefore, cases consider initial power levels varying from full power to zero power. The specific initial power levels that are analyzed are 100%, 70%, 30% and 0% as presented in WCAP-8822.

All cases consider the largest possible break, a double-ended rupture (DER) immediately downstream of the flow restrictor at the outlet of the steam generator. This break conservatively bounds the plant response to any smaller break size. The effective forward break area is limited by the 1.4 ft^2 cross-sectional area of the flow restrictor. The reverse break area is the cross-sectional area of the pipe, which is 4.125 ft^2.

Several single failures can be postulated that would impair the performance of various steamline break protection systems. The single failures either reduce the heat removal capacity of the containment safeguards systems or increase the energy release from the steamline break. The single failures that have been postulated for Ginna are summarized below. The analysis cases separately consider each single failure at each initial power level.

1) **AFW Throttle Valve Failure**

The AFW system includes flow control valves that are designed to throttle the AFW flow from the motor-driven pumps. It is postulated that this valve fails full open instead of throttling as intended. This affects the flowrate from a single motor-driven AFW pump. This failure results in an AFW flowrate to the faulted SG of 265 gpm, versus the design outlet flowrate of $\leq$ 230 gpm.

2) **Main Feedwater Regulating Valve (MFRV) Failure**

The MFRV is a fast-closing (10 second stroke time) valve in the feedwater system that is the preferred (fastest) method for terminating feedwater addition to the faulted SG during a steamline break. If the MFRV on the faulted loop fails open, the back-up main feedwater isolation valve (MFIV), with a 30 second stroke time, is credited to close. The slower closure time creates the possibility of additional pumped feedwater entering the faulted SG. Although the main feedwater pumps trip on a safety injection (SI) signal, the condensate pumps do not trip and can continue to provide pumped flow if the faulted SG depressurizes below approximately 350 psia.

3) **Vital Bus Failure**

One of the postulated failures is that of a vital bus that powers one safeguards train. The main impact is that the active containment heat removal is reduced by 50% with the loss of one train of fan coolers and one containment spray pump. The failure also causes the loss of one train of safety injection.

4) **Diesel Failure**

A diesel failure is postulated only after a loss of offsite power. A loss of offsite power necessitates the start of diesels to power the required safety-related equipment. A diesel failure eliminates one safeguards train, including one train of fan coolers, one containment spray pump, one train of safety injection and one train of motor-driven auxiliary feedwater pumps. This failure is similar to the vital bus failure, above, except there are longer delays because of the time to power equipment from the functioning diesels. In addition, the loss of offsite power has other impacts on the event such as tripping the RCPs and the loss of a motor-driven AFW pump, which would have been powered by the failed diesel.

4.0 Analysis Assumptions

4.1 Protection Logic and Setpoints

The pertinent signals and setpoints that are actuated in these analyses are summarized below.

The first safety injection (SI) signal is generated by a low steamline pressure signal in all cases. The assumed setpoint is 372.7 psia, with a lead/lag of 12/2. The SI signal is credited to cause:

- start of SI pumps,
- reactor trip,
- start of motor-driven auxiliary feedwater pumps,
- closure of MFRVs and MFIVs, and
- trip of main feedwater pumps.

The start of containment fan coolers is credited after the hi containment pressure setpoint of 6 psig is reached. The start of containment sprays is credited after the hi-hi containment pressure setpoint of 33.5 psig is reached.

4.2 Secondary Side Assumptions

This section summarizes the major input assumptions associated with the steam generator, the main feedwater system, the auxiliary feedwater system and the steamline.

Initial Steam Generator Inventory

A high initial steam generator mass is assumed. The initial level corresponds to 60% narrow range span (NRS) at all power levels. This consists of a nominal level of 52% NRS, 4%NRS measurement uncertainty, and 4%NRS measurement bias.

Main Feedwater System

The main feedwater flowrate to the faulted SG is conservatively high to maximize the water mass inventory that will be converted to steam and released from the break. The MFRV on the faulted loop is assumed to quickly open in response to the increased steam flow. The intact loop MFRV is assumed to either be in the nominal position based on the initial power level, or to be closed, which causes the maximum flowrate to the faulted SG. The trip of the main feedwater pumps is credited after an SI signal, with a 2 second delay and a 10 second coastdown. The closure of the faulted loop MFRV is credited after a 2 second delay and a 10 second valve stroke time. The closure of the MFIVs is credited after a 2 second delay and a 30 second valve stroke time.

The feedwater in the unisolable feedline between the MFRV and faulted SG is also considered in the analysis. The hot main feedwater usually reaches saturated conditions as the SG and feedline depressurize. The decrease in density as flashing occurs causes most of the unisolable feedwater to enter the faulted SG. The modeling of the unisolable feedline volume is reduced by the pipe volume that is purged by the auxiliary feedwater, which remains subcooled at pressures down to atmospheric conditions.

Auxiliary Feedwater

The auxiliary feedwater flowrate to the faulted SG is conservatively high to maximize the water mass inventory that will be converted to steam and released from the break. The motor-driven auxiliary feedwater pumps are actuated on an SI signal. The maximum flowrate assumed in the analysis is 235 gpm to each SG. A minimum delay of 25 seconds is credited in most cases due to the safeguards sequencer logic; cases initiated from zero power do not credit a delay.

The turbine-driven AFW pump is assumed to deliver a maximum flowrate of 630 gpm to the faulted SG. The turbine-driven AFW pump is assumed to be actuated due to a low-low SG level signal in both SGs for any case initiated from a power level above 50%. The turbine-driven AFW pump is also actuated due to a loss of offsite power, and is therefore modeled for all diesel failure cases.

Operator action is credited to terminate the auxiliary feedwater flow to the faulted steam generator after 10 minutes.

Unisolable Steamline

The initial steam in the steamline between the break and the steamline nonreturn check valve is included in the mass and energy released from the break.

Quality of the Break Effluent

The quality of the break effluent is assumed to be 1.0, corresponding to saturated steam that is all vapor with no liquid. Although it is expected that there would be a significant quantity of liquid in the break effluent for a full double-ended rupture, the all-vapor assumption conservatively maximizes the energy addition to the containment atmosphere.

4.3 Reactor Coolant System Assumptions

While the mass and energy released from the break is determined from assumptions that have been discussed in the previous section, the rate at which the release occurs is largely controlled by the conditions in the reactor coolant system. The major features of the primary side analysis model are summarized below.

- Continued operation of the reactor coolant pumps maintains a high heat transfer rate to the steam generators. However, the diesel failure cases follow a loss of offsite power, and therefore credit the trip of the RCPs.

- The model includes consideration of the heat that is stored in the RCS metal.

- Reverse heat transfer from the intact steam generator to the RCS coolant is modeled as the temperature in the RCS falls below the steam generator fluid temperature.

- Minimum flowrates are modeled from ECCS injection, to conservatively minimize the amount of boron that provides negative reactivity feedback.

- The assumed NSSS power is 1817 MWt, which includes a maximum pump heat of 10 MWt.

- RCS average temperature is the full-power nominal value of 576.0°F plus an uncertainty of +4.0°F.

- Core residual heat generation is assumed based on the 1979 ANS decay heat plus 2σ model (Ref. 11).

- Conservative core reactivity coefficients (e.g. moderator temperature) corresponding to end-of-cycle conditions with the most reactive rod stuck out of the core are assumed. This maximizes the reactivity feedback effects as the RCS cools down as a result of the steamline break.

- All cases have credited a minimum shutdown margin of 1.3% Δk.

4.4 Containment Assumptions

This section identifies the major input values that are used in the containment response analysis. The assumed initial conditions and the input assumptions associated with the fan coolers and containment sprays are listed in Table 1 (similar to UFSAR Table 6.2-9). The containment heat sink input is provided in Table 2 (similar to UFSAR Table 6.2-10), and the corresponding material properties are listed in Table 3 (similar to UFSAR Table 6.2-11).

Table 1

Initial Containment Conditions, Fan Cooler and Containment Spray Pump Assumptions

Parameter	Value	
Containment net free volume (ft^3)	1,000,000	
Initial containment temperature (°F)	120.0	
Initial containment pressure (psia)	15.7	
Initial relative humidity (%)	20	
Number of fan coolers - All - Vital bus or diesel failure	 4 2	
Hi containment pressure setpoint (psig)	6.0	
Delay (sec) from high containment pressure setpoint to start of fan coolers - With offsite power available - With loss of offsite power	 34.0 44.0	
Containment fan cooler heat removal (BTU/sec per fan cooler) vs. containment temperature (°F)	Temp	Heat Removal
	85	0
	120	398
	220	8839
	240	10375
	260	11911
	280	13446
	286	13907
Number of spray pumps - All - Vital bus or diesel failure	 2 1	
Hi-hi containment pressure setpoint (psig)	33.5	
Delay (sec) from high-high containment pressure setpoint to start of containment sprays: - 2 sprays - 1 spray	 26.8 28.5	
Containment spray flowrate (gpm/spray pump)	1300.0	
RWST/containment spray water temperature (°F)	104.0	

Table 2 Containment Heat Sink Input					
Heat Sink Number	Description	Area (ft^2)	Material	Thickness (inches)	Thickness (ft)
1	Insulated Containment Wall [1]	36285	SS	0.019	0.00158
			Insulation	1.250	0.1042
			Steel	0.375	0.03125
			Concrete	42.000	3.5
2	Uninsulated Containment Wall [1]	12370	Overcoat	0.008	
			Primer	0.002	
			Steel	0.375	0.03125
			Concrete	30.000	2.5
3	Basement Floor [1]	6576	Overcoat	0.005	
			Concrete	24.000	2
			Steel	0.250	0.0208
			Concrete	24.000	2
4	Wet Sump Wall A [1]	8.2	Overcoat	0.004	
			Primer	0.002	
			Steel	0.250	0.0208
			Concrete	36.000	3
5	Dry Sump Wall A [1]	2052.8	Overcoat	0.004	
			Primer	0.002	
			Steel	0.250	0.0208
			Concrete	36.000	3
6	Sump Floors [1]	366	Overcoat	0.005	
			Concrete	24.000	2
			Steel	0.250	0.0208
			Concrete	12.000	1
7	Walls of Sump B [1]	189	Overcoat	0.005	
			Concrete	24.000	2
			Steel	0.250	0.0208
			Concrete	12.000	1
8	Outer Refueling Cavity Wall	6132	Overcoat	0.005	
			Concrete	35.280	2.94
9	Inner Refueling Cavity Wall [1]	5609	SS	0.250	0.0208
			Concrete	24.000	2
10	Bottom Refueling Cavity [1]	1143	SS	0.250	0.0208
			Concrete	48.000	4
11	Loop Compartments	18846	Overcoat	0.005	
			Concrete	16.938	1.4115
12	Floor of Intermediate Level	9672	Overcoat	0.005	
			Concrete	3.000	0.25

Table 2 Containment Heat Sink Input					
Heat Sink Number	Description	Area (ft^2)	Material	Thickness (inches)	Thickness (ft)
13	Operating Deck	15570	Overcoat	0.005	
			Concrete	12.000	1
14	Thick Crane Structure	7225	Overcoat	0.004	
			Primer	0.002	
			Steel	0.750	0.0625
15	Crane Structure	3374	Overcoat	0.004	
			Primer	0.002	
			Steel	0.415	0.03455
16	I-Beam	7678	Overcoat	0.004	
			Primer	0.002	
			Steel	0.260	0.0217
17	Thick I-Beam	5536	Overcoat	0.004	
			Primer	0.002	
			Steel	0.703	0.0586
18	Crane Support	342	Overcoat	0.004	
			Primer	0.002	
			Steel	2.000	0.16667
19	Crane Beams	236	Overcoat	0.004	
			Primer	0.002	
			Steel	1.440	0.12
20	Grating and Misc	14000	Overcoat	0.004	
			Primer	0.002	
			Steel	0.0625	0.005208

Note:
1. The air gaps between concrete and steel and between insulation and steel are modeled.

Table 3 Material Properties Table for Containment Heat Sinks		
Material	Conductivity (BTU/hr-ft-°F)	Volumetric Heat Capacity (BTU/ft^3-°F)
Concrete	0.81	31.5
Carbon Steel	28.0	54.4
Insulation	0.0208	1.11
Stainless Steel	8.8	54.6
Organic Coating	0.1	20.0
Inorganic Primer	1.0	20.0

5.0 Steamline Break Containment Response Results

Sixteen steamline break cases were analyzed varying the initial power level and the assumed single failure. The mass and energy release from the break was calculated using the RETRAN code, while the containment pressure response was determined with the GOTHIC code. The analysis included the effects of the extended power uprate to 1817 MWt, a decrease in the shutdown margin to 1.3%, and the benefit of the modification to automatically close the feedwater isolation valves on an SI signal.

The current limiting steamline break containment pressure case, as documented in FSAR Section 6.2.1.2.3 (Rev. 18), is a full power double-ended break with a MFRV failure. With the faulted loop MFRV assumed to fail open, this analysis models feedwater isolation due to the closure of the main feedwater pump discharge valves with an 80-second stroke time. The total water mass addition from the main feedwater system is over 100,000 lbm to the faulted SG. A key element in limiting the break release rate is crediting the 2.4% shutdown margin, which prevents a post-trip return-to-power.

Figure 1 and Figure 2 compare this limiting case from the FSAR to the new analysis which has credited feedwater isolation due to the automatically actuated MFIV with a stroke time of 30 seconds. In the first 120 seconds, there is a higher release rate and a higher containment pressure due to the EPU effects and a lower shutdown margin (1.3%). However, the MFIV closure limits the main feedwater addition to less than 20,000 lbm, which substantially improves the longer-term containment pressure transient. The peak containment pressure is reduced from 59.8 psig to 50.8 psig.

With the modified MFIV, an MFRV failure is no longer the limiting single failure for the steamline break containment response analysis. The limiting case definition changes to a vital bus failure with an initial power level of 70%. The sequence of events for this limiting case is listed in Table 4. The break flowrate is shown in Figure 3, the break enthalpy is in Figure 4, and the containment pressure transient is in Figure 5. The peak containment pressure is 59.4 psig, which is below the containment design pressure of 60.0 psig.

Table 4
Sequence of Events
Steamline Break, Vital Bus Failure, 70% Power

Event	Time (sec)
Break occurs	0
Low steamline pressure SI setpoint reached	< 0.05
Hi containment pressure setpoint reached	2.0
Reactor trip	2.0
Main feedwater pumps trip	2.0
MFRV closes	12.0
Auxiliary feedwater starts	25.0
Containment fan coolers start	36.0
Hi-hi containment pressure setpoint reached	40.0
Containment sprays start	68.5
AFW terminated to faulted SG	600
Peak containment pressure occurs	612
Break releases stop	716

6.0 Conclusion

The steamline break mass and energy release and containment response analysis has been done to show the effect of the automatically-actuated MFIVs as a back-up feedwater isolation method if the faulted loop MFRV fails open. This plant modification is a benefit to the containment pressure response, lowering the peak containment pressure by 9 psi for this postulated single failure.

The analysis also considered more limiting conditions associated with the EPU and a reduction in the minimum required shutdown margin at end of cycle conditions for a spectrum of cases. With the MFIV plant modification, the limiting case definition changes to a double-ended rupture initiated at 70% power with a vital bus failure. The peak containment pressure is 59.4 psig, which is acceptable because it is below the containment design pressure of 60.0 psig.

Figure 1

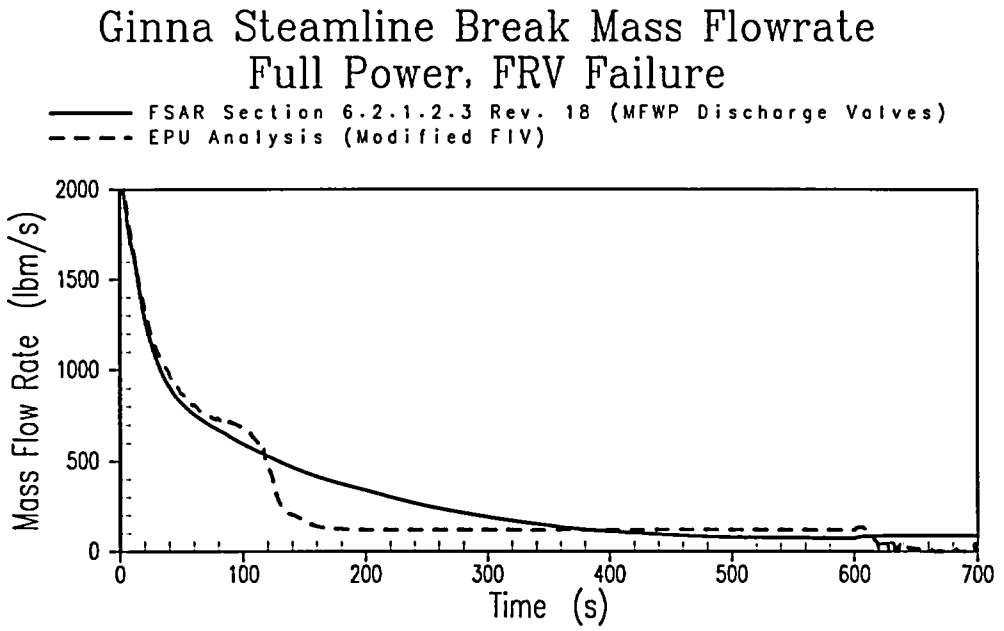


Figure 2

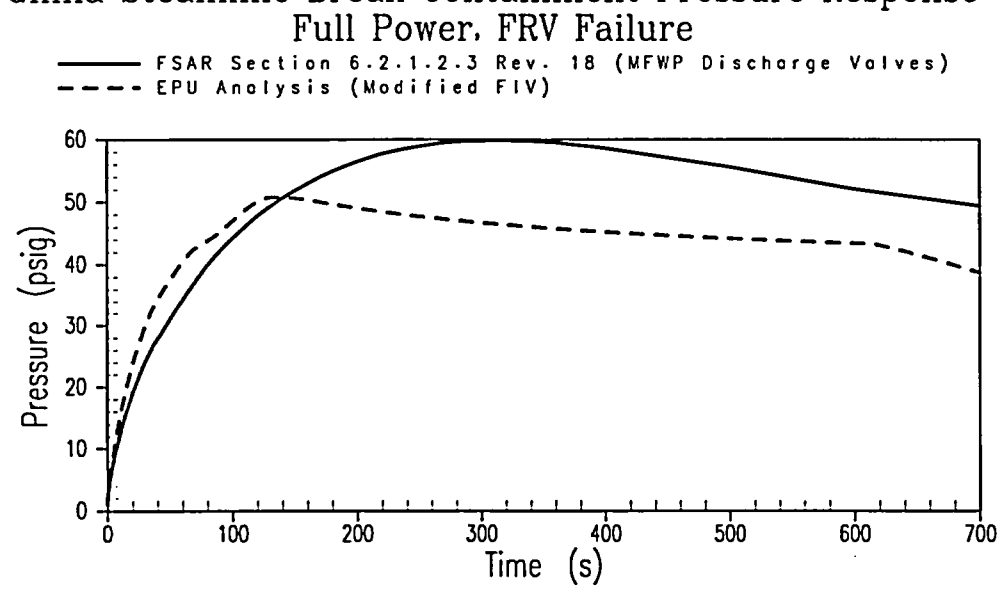


Figure 3

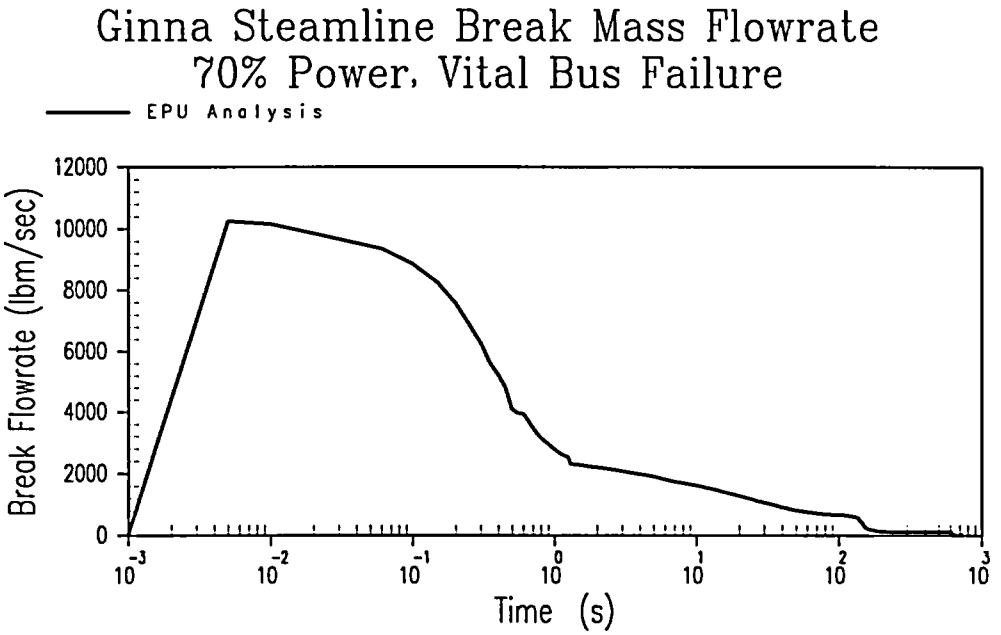


Figure 4

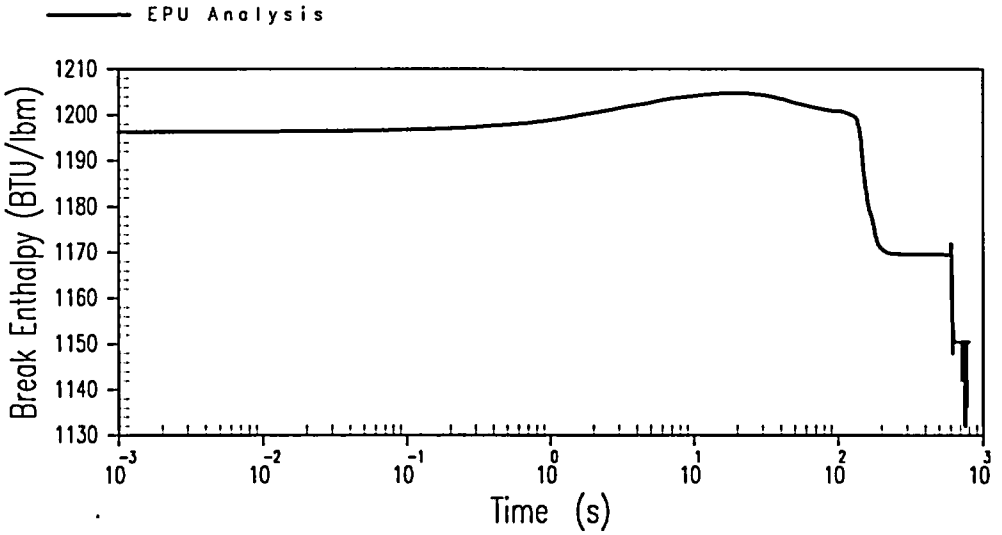
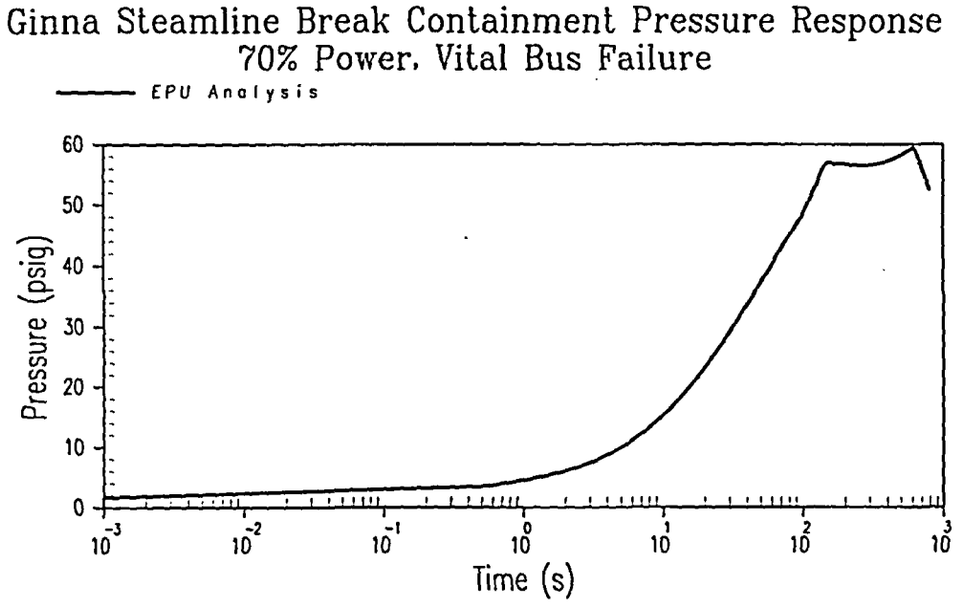


Figure 5


Ginna Steamline Break Containment Pressure Response
70% Power, Vital Bus Failure
EPU Analysis
Pressure (psig)
60
50
40
30
20
10
0
10^{-3}
10^{-2}
10^{-1}
10^{0}
10^{1}
10^{2}
10^{3}
Time (s)

7.0 References

1. Letter from Cecil O. Thomas (NRC), 'Acceptance for Referencing of Licensing Topical Report WCAP-8821(P)/8859(NP), "TRANFLO Steam Generator Code Description", and WCAP-8822(P)/8860(NP), "Mass and Energy Release Following a Steam Line Rupture," August 1983.

2. Land, R.E., "Mass and Energy Releases Following a Steam Line Rupture," WCAP-8822 (Proprietary), WCAP-8860 (Non-Proprietary), September 1976.

3. Burnett, T.W.T., et al. "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Non-Proprietary), April 1984.

4. Osborne, M. P. and D. S. Love, "Mass and Energy Releases Following a Steam Line Rupture, Supplement 1 – Calculations of Steam Superheat in Mass/Energy Releases Following a Steamline Rupture," WCAP-8822-S1-P-A (Proprietary), September 1986.

5. Huegel, D. S, et al. "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," WCAP-14882-P-A (Proprietary), April 1999.

6. NAI 8907-06, Revision 13, "GOTHIC Containment Analysis Package Technical Manual," Version 7.1, January 2003.

7. NAI 8907-09, Revision 7, "GOTHIC Containment Analysis Package Qualification Report," Verstion 7.1, January 2003.

8. Bordelon, F. M., and E. T. Murphy, "Containment Pressure Analysis Code (COCO)," WCAP-8327 (Proprietary), WCAP-8326 (Non-Proprietary), July 1974.

9. NRC Letter from Anthony C. McMurtray (NRC) to Thomas Coutu (NMC), Enclosure 2, Safety Evaluation, September 29, 2003.

10. NAI 8907-02, Revision 14, "GOTHIC Containment Analysis Package User Manual," Version 7.1, January 2003.

11. ANSI/ANS-5. 1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

ENCLOSURE 6
R.E. Ginna Nuclear Power Plant

Steam Line Break at Hot Zero Power

SteamLine Break at Hot Zero Power

1.0 Introduction

The steam release from a major rupture of a main steam pipe will result in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of reactor coolant temperature and pressure. In the presence of a negative moderator temperature coefficient (MTC), the cooldown results in a positive reactivity insertion and subsequent reduction in core shutdown margin. If the most-reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam pipe rupture is a concern primarily because of the high-power peaking factors that would exist assuming the most-reactive RCCA is stuck in its fully withdrawn position. The core is ultimately shut down by boric acid injection delivered by the ECCS.

The major rupture of a main steam pipe is the most-limiting cooldown transient. It is analyzed at HZP conditions with no decay heat (decay heat would retard the cooldown, thus reducing the return to power). A detailed discussion of this transient with the most limiting break size is presented below.

The primary design features which provide protection for steam pipe ruptures are:

- Actuation of the SI system from any of the following:
  - Two-out-of-three pressurizer low-pressure signals.
  - Two-out-of-three low-pressure signals in any steam line.
  - Two-out-of-three high-containment pressure signals.
- If the reactor trip breakers are closed, reactor trip can be actuated from overpower neutron flux, overpower delta T (OPΔT), or upon actuation of the SI system.
- Redundant isolation of the main feedwater lines to prevent sustained high-feedwater flow that will cause additional cooldown. In addition to the normal control action which will close the main feedwater control valves, an SI signal will also rapidly close all feedwater control valves as well as the feedwater isolation valves.
- Trip of the fast-acting main steamline isolation valves (MSIVs), on the following:
  - Two-out-of-three high containment pressure signals.
  - One-out-of-two high-high steam flow signals in a steam line in coincidence with any safety injection signal.
  - One-out-of-two high-steam flow signals in a steam line in coincidence with two-out-of-four indications of low-reactor coolant T_{avg} and any SI signal.

Each steam line is provided with a main steam isolation valve which isolates flow in the forward direction, and a main steam non-return valve, which isolates flow in the reverse direction. Thus, even with a single failure of any valve, no more than one steam generator can blow down, no matter where the break is postulated. The unaffected steam generator is still available for dissipation of decay heat after the initial transient is over.

Following blowdown of the faulted steam generator, the unit can be brought to a stabilized hot-standby condition through control of the auxiliary feedwater (AFW) flow and SI flow as described by plant operating procedures. The operating procedures would call for operator action to limit RCS pressure and pressurizer level by terminating SI flow and to control steam generator level and RCS coolant temperature using the auxiliary feedwater system (AFWS).

2.0 Input Parameters, Assumptions, and Acceptance Criteria

The following summarizes the major input parameters and/or assumptions used in the main steam line rupture event:

- HZP conditions were modeled with two loops in service with and without offsite power available. A case with one loop in service with offsite power available was also modeled.

- For Ginna, a 1.4 ft^2 break was analyzed for the Babcock and Wilcox (BWI) steam generators, since they are designed with a flow restrictor built into the steam exit nozzle. The assumed steam generator tube plugging level was 0%.

- All control rods were inserted except the most reactive RCCA, which was assumed to be stuck out of the core.

- The shutdown margin was 1.30% $\Delta k/k$ and 1.80% $\Delta k/k$ for the two-loop and one-loop operation cases, respectively.

3.0 Description of Analyses and Evaluations

A detailed analysis using the RETRAN (Reference 1) computer code was performed in order to determine the plant transient conditions following a main steam line break. The code models the core neutron kinetics, RCS, pressurizer, steam generators, SI system and the AFWS; and computes pertinent variables, including the core heat flux, RCS temperature, and pressure. A conservative selection of those conditions were then used to develop core models which provide input to the detailed thermal and hydraulic digital computer code, VIPRE (Reference 2), to determine if the DNB design basis is met.

4.0 Steamline Break at HZP Results

For Ginna, the most limiting main steamline rupture at HZP case is the case with two-loops in service in which offsite power is assumed to be available since the steam generator inventory is highest and the reactor coolant pumps are available to circulate RCS flow.

The calculated sequence of events for the most limiting case is shown in Table 1. Figures 1 through 4 show the transient results for the most limiting case for Ginna. These figures show transient results following a 1.4 ft^2 main steamline rupture at initial no-load conditions with offsite power available. Since offsite power is assumed available, there is full reactor coolant flow.

Should the core be critical at near zero power when the rupture occurs, the initiation of SI via a low-steam line pressure signal will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the main steam isolation valves in conjunction with the main steam non-return valves.

As shown in Figure 3 the core attains criticality with the RCCAs inserted (i.e., with the plant shutdown assuming one stuck RCCA) before boron solution from the ECCS enters the RCS.

A DNB analysis was performed for the limiting point in the transient which determined that the DNB design basis is met. The peak heat flux (13.3%) and minimum DNBR (2.58) occur approximately 54 seconds after the break occurs. The DNBR Limit used for this event is 1.566. Primary and secondary pressure limits are not challenged because primary and secondary pressures decrease from their initial values during the transient.

5.0 Steamline Break at HZP Conclusions

The only criterion that could be challenged during this event is the one that states that the critical heat flux should not be exceeded. The analysis demonstrated that this criterion was met by showing that the minimum DNBR did not go below the limit value at any time during the transient.

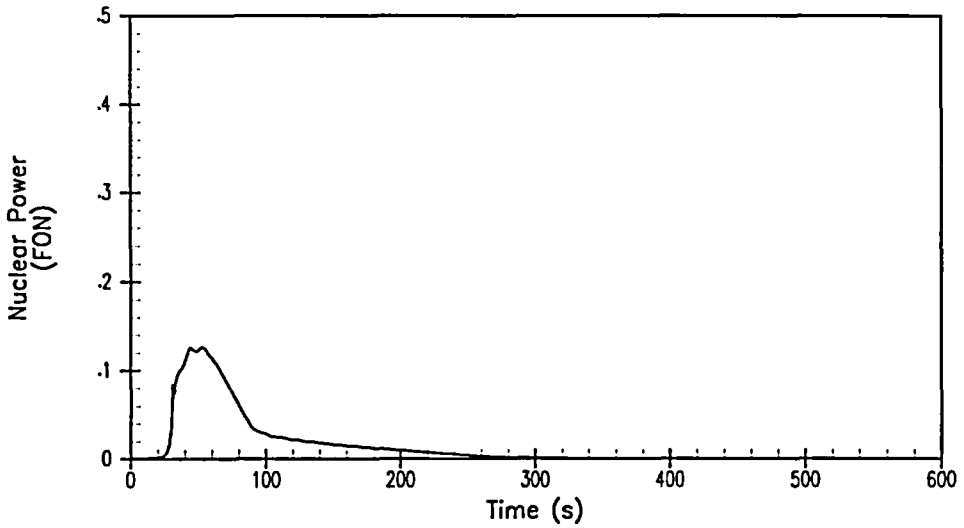
The results of the major rupture of a main steam pipe event indicate that the DNB design basis is met. The calculated minimum DNBR is 2.58 compared to a limit of 1.566. Primary and secondary pressure limits are not challenged because primary and secondary pressures decrease from their initial values during the transient. Therefore, this event does not adversely affect the core or the RCS, and all applicable acceptance criteria are met.

6.0 Steamline Break at HZP References

1. WCAP-14882-P-A (Proprietary), April 1999 and WCAP-15234-A (Nonproprietary), *RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses*, Huegel, D. S., et al., May 1999.

2. WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Nonproprietary), *VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis*, Sung, Y. X. et al., October 1999.

Table 1 Ginna Station Time Sequence of Events – Steamline Break at Hot Zero Power		
Case	**Event**	**Time (sec)**
Reactor at HZP with Offsite Power Available (Unisolatable Steam Release Paths Case)	Double-Ended Guillotine Break Occurs	0.0
	Low Steam Pressure SI System Actuation Setpoint Reached	1.4
	MSIVs Closed 7 Seconds After SI System Actuation Signal	8.4
	High-Head SI Pump At Rated Speed 12 Seconds After SI System Actuation Signal	15.4
	Reactor Becomes Critical	22.7
	Main Feedwater Flow Isolated 32 Seconds After SI System Actuation Signal	33.4
	Power Reaches Maximum Level	53.0
	Time of Minimum DNBR	54.2
	Reactor Returns Subcritical	67.5



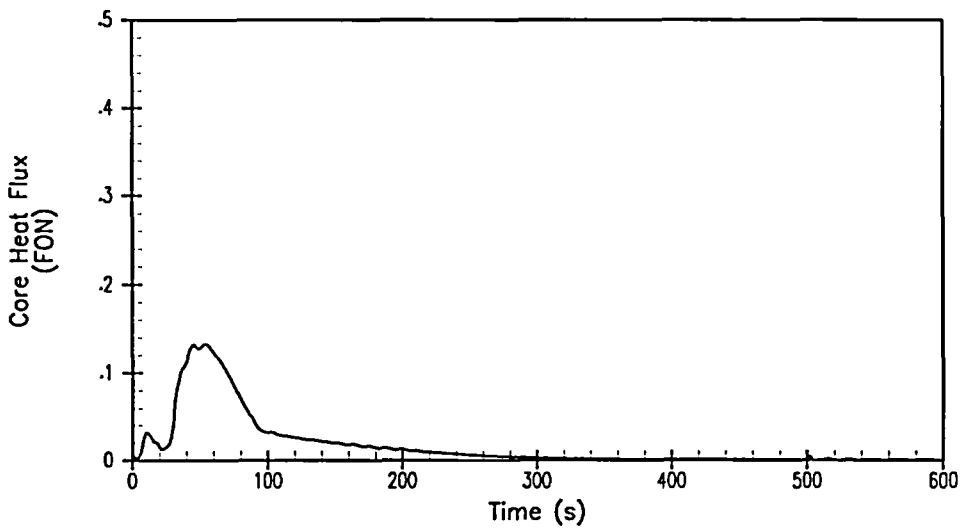


Figure 1
Ginna Station
Steamline Break at Hot Zero Power – 1.4 ft^2 Break
(with Offsite Power Available, Unisolatable Steam Paths)
Nuclear Power, and Core Heat Flux vs. Time

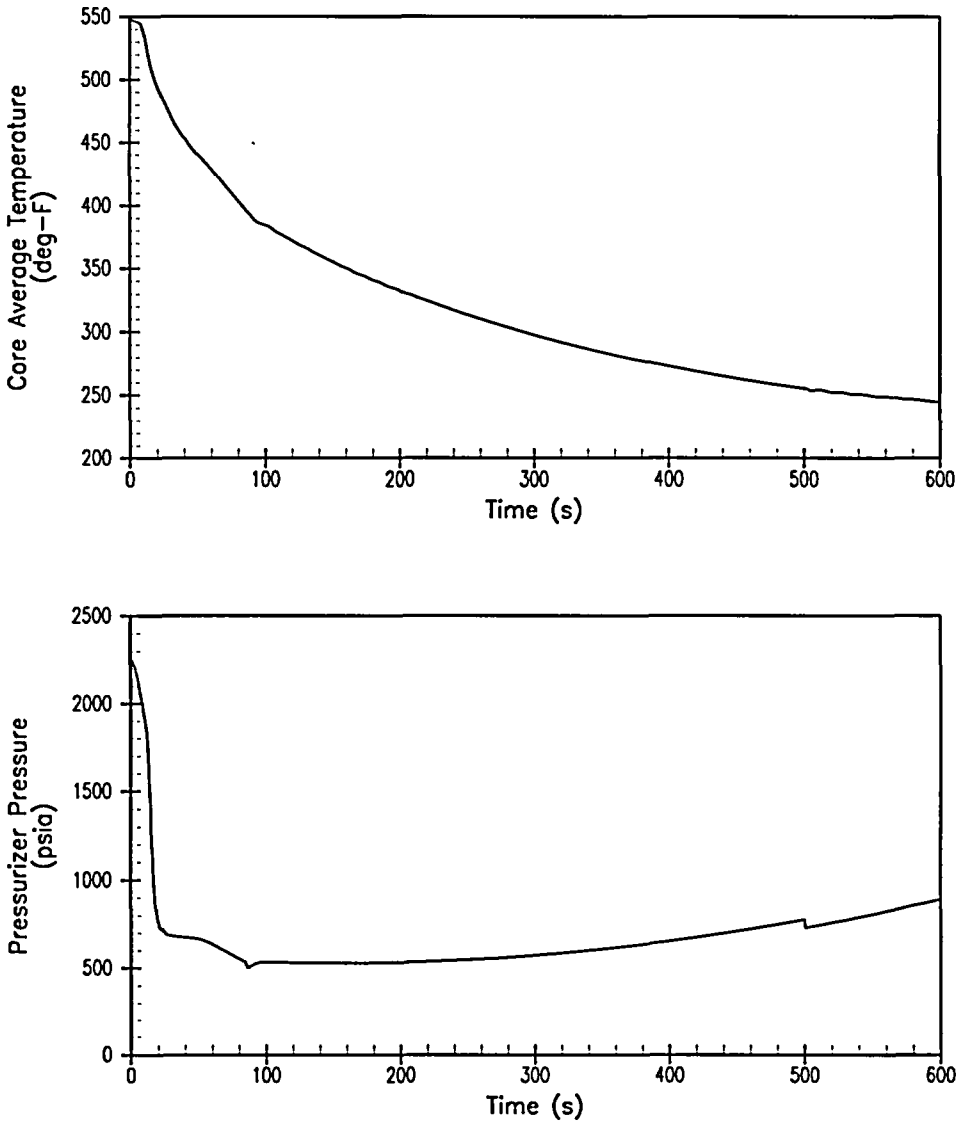
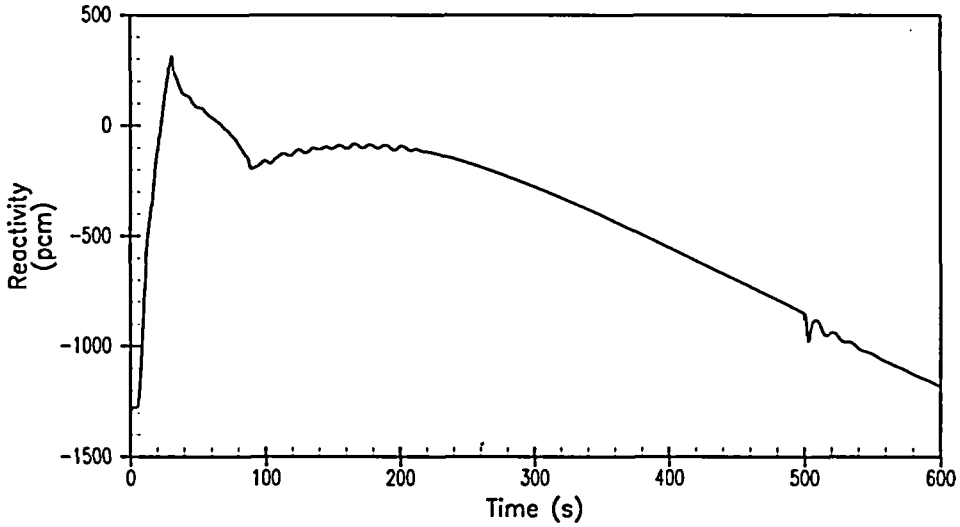


Figure 2
Ginna Station
Steamline Break at Hot Zero Power – 1.4 ft^2 Break
(with Offsite Power Available, Unisolatable Steam Paths)
Core Average Temperature, and Pressurizer Pressure vs. Time



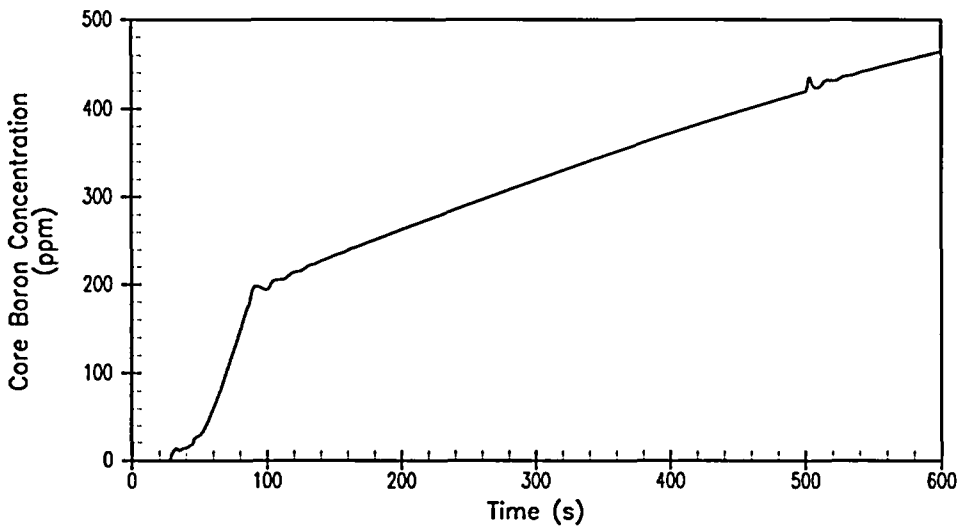
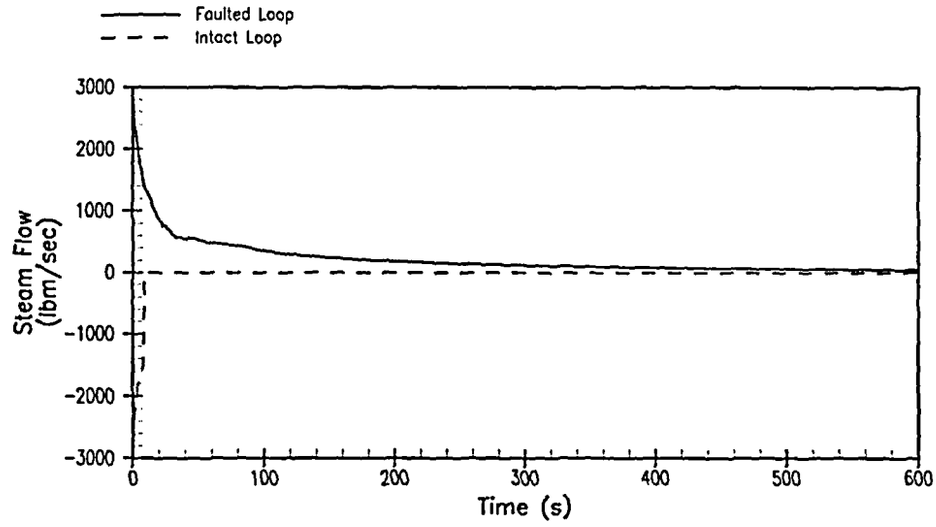


Figure 3
Ginna Station
Steamline Break at Hot Zero Power – 1.4 ft^2 Break
(with Offsite Power Available, Unisolatable Steam Paths)
Reactivity, and Core Boron Concentration vs. Time



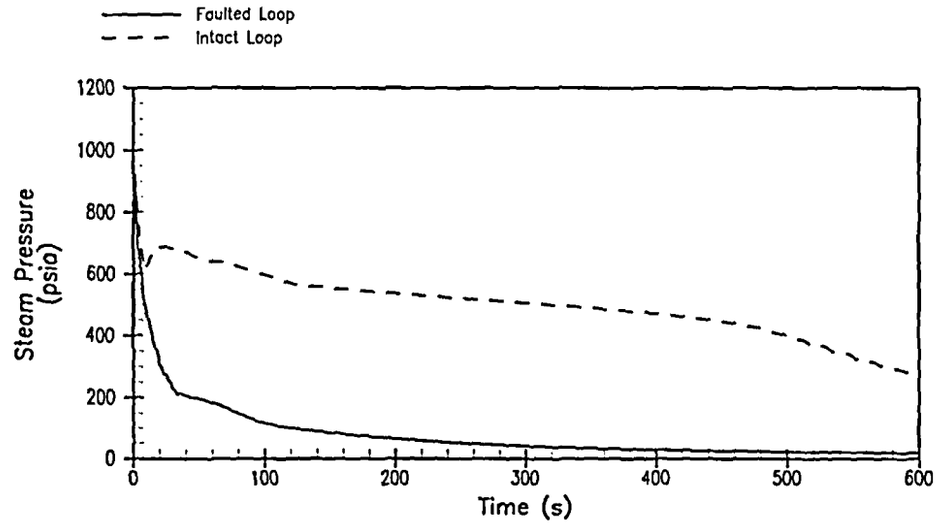


Figure 4
Ginna Station
Steamline Break at Hot Zero Power – 1.4 ft^2 Break
(with Offsite Power Available, Unisolatable Steam Paths)
Steam Flow, and Steam Pressure vs. Time

ENCLOSURE 7
R.E. Ginna Nuclear Power Plant

List of Regulatory Commitments

The following table identifies those actions committed to by R.E. Ginna Nuclear Power Plant, LLC in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

REGULATORY COMMITMENT	DUE DATE
Modify existing MFIVs 3994 and 3995.	Prior to startup from the fall 2006 refueling outage.
Once approved, the amendment will be implemented.	Prior to startup from the fall 2006 refueling outage.
This change in the analysis incorporates additional margin for instrument uncertainty and will be addressed separately in the planned license amendment request for the extended power uprate.	Submittal of extended power uprate license amendment is currently planned by June 30, 2005.