WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J. Garrett
Vice President, Engineering

September 12, 2007
ET 07-0040

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

Reference: Letter ET 07-0004, dated March 14, 2007, from T. J. Garrett, WCNOC, to USNRC

Subject: Docket No. 50-482: Response to Request for Additional Information Relating to Replacement of the Main Steam and Feedwater Isolation Valves and Controls

Gentlemen:

The Reference provided a license amendment request that proposed revisions to Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," TS 3.7.2, "Main Steam Isolation Valves (MSIVs)," and TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)." The Reference proposed changes to these specifications based on a planned modification to replace the MSIVs and associated actuators, MFIVs and associated actuators, and replacement of the Main Steam and Feedwater Isolation System (MSFIS) controls.

The NRC provided by electronic mail on July 2, 2007, a request for additional information related to the steam generator tube rupture dose consequence analysis that is referred to in Section 4.11 of Attachment I of the Reference. Wolf Creek Nuclear Operating Corporation (WCNOC) provided a response to the questions by electronic mail on July 19, 2007. Attachment I to this letter provides the response to the request for additional information.

The NRC provided by electronic mail on July 31, 2007, an additional request for additional information that was discussed in a teleconference on August 3, 2007. Attachment II to this letter provides the response to the request for additional information.

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The additional information provided in the Attachments do not impact the conclusions of the No Significant Hazards Consideration provided in Reference 1. In accordance with 10 CFR 50.91, a copy of this submittal is being provided to the designated Kansas State official.

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-4084, or Mr. Kevin Moles at (620) 364-4126.

Sincerely,

Terry J. Garrett

TJG/rlt

Attachments

cc: J. N. Donohew (NRC), w/a
V. G. Gaddy (NRC), w/a
B. S. Mallett (NRC), w/a
Senior Resident Inspector (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Terry J. Garrett, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By____________________
Terry J. Garrett
Vice President Engineering

SUBSCRIBED and sworn to before me this 12th day of September 2007.

GAYLE SHEPHEARD
Notary Public - State of Kansas
My Appt. Expires 7/24/2011

Gayle Shepheard
Notary Public

Expiration Date 7/24/2011

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

The NRC provided by electronic mail on July 2, 2007, a request for additional information related to the steam generator tube rupture dose consequence analysis that is referred to in Section 4.11 of Attachment I of the Wolf Creek Nuclear Operating Corporation (WCNOC) letter ET 07-0004. Letter ET 07-0004 provided a license amendment request that proposed revisions to Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," TS 3.7.2, "Main Steam Isolation Valves (MSIVs)," and TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)" based on a planned modification to replace the MSIVs and associated actuators, MFIVs and associated actuators, and replacement of the Main Steam and Feedwater Isolation System (MSFIS) controls. WCNOC provided a response to the questions by electronic mail on July 19, 2007. Provided below are responses to the questions in the request for additional information.

1. Describe each parameter used in the summary of the steam generator tube rupture (SGTR) dose consequence analysis that is referred to in Section 4.11 of Attachment I to the License Amendment Request (LAR) dated March 14, 2007 (ET 07-0004). For each parameter, indicate whether or not the parameter represents a change to the value in the current licensing basis SGTR analysis.

Response: The seven assumptions and parameters referred to in Section 4.11 of Attachment I to letter ET 07-0004 are discussed as follows for the limiting SGTR w/forced overfill dose consequence analysis:

1. The primary-to-secondary break flow discharge and releases to the environment from the affected and intact steam generators (SG) are obtained from RETRAN analyses. The differences between the calculated break flow from the reanalysis and the break flow from the analysis of record are due primarily to the different MSIV and MFIV closure characteristics. Integrated break flow and releases are as follows:

Parameter	Analysis of Record	Reanalysis
Integrated break flow to the end of the transient, lbs	196,930	195,371
Faulted SG steam release to atmosphere, 0-2 hours, lbs	145,650	140,406
Faulted SG steam release to atmosphere, until shutdown cooling is in operation and SG release terminated, lbs	131,760	120,960
Intact SG steam release to atmosphere, 0-2 hours, lbs	298,783	309,069
Intact SG steam release to atmosphere, until shutdown cooling is in operation and SG release terminated, lbs	63,360	299,520

2. All iodine contained in the fraction of reactor coolant flashing to steam upon reaching the secondary side is assumed released to the steam phase, with no credit for scrubbing. This does not represent a change to the value in the current licensing basis SGTR analysis of record.

3. A one gpm primary-to-secondary leak rate is assumed to occur to the unaffected SGs, throughout the accident sequence. This assumption is conservative with respect to the Technical Specification limit of 150 gpd (.104 gpm) primary-to-secondary leakage through any one SG. This represents a change to the value of one gpm primary-to-secondary leak rate assumed to occur to each of the unaffected SGs (3 gpm total) in the current licensing basis SGTR analysis of record.

4. All noble gas activity in the reactor coolant that is transported to the secondary system via the tube rupture and the primary-to-secondary leakage is assumed released to the atmosphere. This does not represent a change to the value in the current licensing basis SGTR analysis of record.

5. The iodine partition fraction between the liquid and steam in the SG is assumed to be 0.01. This does not represent a change to the value in the current licensing basis SGTR analysis of record.

6. The steam releases were assumed to continue from the intact SGs for a period of 8 hours until long term cooling operation was achieved. This is a departure from the current analysis of record in which long term cooling operation was achieved within four hours.

7. Radioactive decay prior to the release of activity is considered. No decay during transit or ground deposition is considered. This does not represent a change to the value in the current licensing basis SGTR analysis of record.

In addition, as discussed in Section 4.11 of Attachment I to letter ET 07-0004, three elements of the analysis of record methodology were changed:

1. The reanalysis uses thyroid dose conversion factors for inhalation of radioactive material based on the data provided in Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion." This is a departure from the more conservative dose conversion factor values listed in Regulatory Guide 1.109, which were used in the current analysis of record.

2. The effective dose conversion factors, provided in Table III.1 of EPA Federal Guidance Report No. 12 1993, "External Exposure to Radionuclides in Air, Water, and Soil," are used to calculate the whole body doses. This deviates from the current analysis of record, which used the more conservative dose conversion factor values listed in Regulatory Guide 1.109.

3. The reanalysis uses a factor of 335 for the accident initiated iodine spike release rate, which is a departure from the more conservative factor of 500 modeled in the current analysis of record. Note: The pre-accident iodine spike primary coolant iodine concentration maximum value of 60 μCi/gm DE I-131 was used in the reanalysis, consistent with the analysis of record.

The engineering calculation AN-07-003, "Wolf Creek Steam Generator Tube Rupture with Stuck Open ARV and with Forced Overfill Radiological Consequence Analyses, with RADTRAD, to Support the MSIV/MFIV Replacement Project (DCP #09952)," Revision 0, dated April 30, 2007, is the calculation supporting the SGTR dose consequence analysis described in Section 4.11 of Attachment I of letter ET 07-0004. An electronic copy of the calculation was provided by electronic mail on July 19, 2007. Subsequently, the NRC determined that the calculation was not required to be placed on the WCNOC docket and the calculation is available for NRC review as needed.

2. Describe the impact of the proposed change on the SGTR dose consequence analysis on control room habitability.

Response: As presented below, the calculated offsite low population zone and site boundary radiological consequences, for both the pre-accident and concurrent iodine spike case reanalysis, are exceeded by or similar to the radiological consequences of the current limiting SGTR w/forced overfill analysis of record. Integrated break flow changes due to the MSIV and MFIV valve and actuator replacement were minimal. The increase in the calculated radiological consequences was substantially due to use of a bounding flashing fraction in the reanalysis, compared with a variable flashing fraction based upon primary and secondary conditions in the analysis of record. These changes were mitigated by the change in dose conversion factors used in the analysis. The accident-initiated iodine spike case radiological consequences were further reduced by the change from a spike factor of 500 to a spike factor of 335.

These calculated offsite radiological consequences provide assurance that the control room habitability radiological consequences due to the proposed change would remain bounded by the current LBLOCA radiological consequence analysis of record.

Radiological Consequences of a Steam Generator Tube Rupture

Pre-accident Iodine Spike Case		
	Analysis of Record*	**Reanalysis***
Exclusion Area Boundary		
Thyroid	57.12	51.77
Whole Body	0.189	0.226
Low Population Zone		
Thyroid	7.92	7.19
Whole Body	0.026	0.034
Concurrent Iodine Spike Case		
	Analysis of Record*	**Reanalysis***
Exclusion Area Boundary		
Thyroid	22.50	22.80
Whole Body	0.195	0.132
Low Population Zone		
Thyroid	3.13	4.83
Whole Body	0.027	0.025

* Dose (rem)

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

The NRC provided by electronic mail on July 31, 2007, an additional request for additional information related to Wolf Creek Nuclear Operating Corporation (WCNOC) letter ET 07-0004. The request for additional information was discussed in a teleconference between the NRC staff and Wolf Creek Nuclear Operating Corporation (WCNOC) personnel on August 3, 2007. Letter ET 07-0004 provided a license amendment request that proposed revisions to Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," TS 3.7.2, "Main Steam Isolation Valves (MSIVs)," and TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)" based on a planned modification to replace the MSIVs and associated actuators, MFIVs and associated actuators, and replacement of the Main Steam and Feedwater Isolation System (MSFIS) controls. WCNOC provided a response to the questions by electronic mail on July 19, 2007. Provided below are responses to the questions in the request for additional information.

1. In Section 4.7, page 75 of 133, there is the statement under the heading "MSIV and MFIV Closure Time" states that a MSIV and MFIV closure time delay of 15 seconds was conservatively assumed in the analyses of main steam line break events. In Section 4.0, page 12 of 133, there are the statements that (1) the closure time for the MFIV varies from approximately six seconds to approximately 50 seconds, for a system pressure ranging from 1100 psig to 0 psig and (2) the closure time for MSIV varies from approximately six seconds to approximately 33 seconds, for a system pressure ranging from 1100 psig to 0 psig. It is also stated that the assumed closure time of 15 second is based on steam generator (SG) pressure expected to remain higher than 400 psig for all affected design bases accidents when the valves are approaching the closed position.

   It is understood that the MSIV and MFIV closure time is a critical parameter that establishes the mass and energy release inside the containment and which affects the peak containment pressure and temperature. Therefore, the staff requests additional information for the following items:

   (a) What are MSIV and MFIV equipment design data that show that their maximum closure time is 15 seconds at a system pressure of 400 psig. How were the equipment design data validated?

   (b) What is the basis for the expected SG pressure to be higher than 400 psig during any accident condition when the closure of MSIVs and MFIVs are required to be no more than 15 seconds.

   (c) For the new replacement MSIVs and MFIVs, discuss the assurance that the valve closure time would not degrade below that needed for the accident analyses in the time interval between the times to test the valves within the inservice testing program.

   (d) Both the proposed curves of acceptable valve closure times for the new MSIVs and MFIVs are in terms of the SG pressure in psig. Since the system medium for the MFIVs is the feedwater, how is the feedwater pressure converted to SG pressure to determine the acceptable MFIV closure time.

Response:

(a) The design specifications, M-628(Q), "Main Steam Stop Valves and Actuators for Containment Isolation to ASME Section III for the Wolf Creek Generating Station," and M-630(Q), "Main Feedwater Stop Valves and Actuators for Containment Isolation to ASME Section III for the Wolf Creek Generating Station," require the closure time to be no greater than 5 seconds from receipt of signal under the most adverse flow condition, which is taken to occur at normal operating pressure and temperature. The requirement to close within 15 seconds at a system pressure of 400 psig comes from the accident analyses, and is used as a bounding condition to produce the closure time acceptance curves.

Each valve actuator was functionally tested during the manufacturing process. The functional tests included characterization of the closure time as a function of system pressure. The test data from the valve vendor has been reviewed and shows that the actuators are capable of closing the valves within the time required by the specifications, and meet the closure time needed to comply with the accident analysis bounding condition.

(b) As described in Section 4.0 of Attachment I to ET 07-0004, a conservative yet bounding closure time of 15 seconds has been assumed for analysis purposes. This closure time was confirmed, based on the transient responses for all affected design basis accidents showing steam generator pressures remaining higher than 400 psig when the valves are approaching the closed position. The table below presents the steam generator pressures at the isolation of feedwater and steamline for all sixteen cases.

Time Sequence of Events for the Steamline Break Mass and Energy Releases to Containment

Case	SI Signal	Steamline Isolation Signal	SI Actuation (sec)	Feedwater Isolation (sec)	Steam Generator Pressure (psia)*	Steamline Isolation (sec)	Steam Generator Pressure (psia)
1	LSP	LSP	1.389	18.389	567.6	18.389	567.6
2	LSP	LSP	1.991	18.991	959.0	18.991	959.0
3	Hi-1 Cont P	Hi-2 Cont P	18.7	35.7	979.9	86.7	774.9
4	LSP	LSP	1.209	18.209	547.0	18.209	547.0
5	LSP	LSP	2.800	19.800	941.1	19.800	941.1
6	Hi-1 Cont P	Hi-2 Cont P	16.8	33.8	937.0	84.7	741.4
7	LSP	LSP	1.133	18.133	535.6	18.133	535.6
8	Hi-1 Cont P	Hi-2 Cont P	14.7	31.7	907.4	79.5	710.4
9	Hi-1 Cont P	Hi-2 Cont P	16.7	33.7	921.8	89.3	722.7
10	LSP	LSP	1.115	18.115	533.7	18.115	533.7
11	Hi-1 Cont P	Hi-2 Cont P	18.8	35.8	941.0	125.5	701.8

Case	SI Signal	Steamline Isolation Signal	SI Actuation (sec)	Feedwater Isolation (sec)	Steam Generator Pressure (psia)*	Steamline Isolation (sec)	Steam Generator Pressure (psia)
12	Hi-1 Cont P	Hi-2 Cont P	19.1	36.1	933.0	108.2	719.6
13	LSP	LSP	1.168	18.168	547.9	18.168	547.9
14	Hi-1 Cont P	Hi-2 Cont P	29.7	46.7	989.1	219.9	744.7
15	Hi-1 Cont P	Hi-2 Cont P	30.7	47.7	952.1	192.7	745.3
16	Hi-1 Cont P	Hi-2 Cont P	30.7	47.7	953.6	192.7	745.4

* Feedwater pressures are not part of the computer output, however, they are expected to be higher than the steam generator pressures.

(c) The replacement MSIVs and MFIVs, like the existing MSIVs and MFIVs, are parallel slide gate valves. Since the valve type is not being changed, replacement of the bodies will not introduce any new failure or degradation mechanisms. Therefore heretofore unidentified degradation of the valve performance from the body replacement is unlikely.

The valve actuator type is being changed from a hydraulic actuator to a system-medium actuator. The system-medium actuator uses fewer active components and subsystems than the existing hydraulic actuator, which reduces the possibility of actuator failure or degraded performance. The system-medium actuators have been designed to use graphite or metallic seals rather than elastomeric seals (i.e. O-rings). Graphite and metallic seals are dimensionally stable, and have a very long environmentally qualified lifespan.

The system-medium actuator has been in service in nuclear power plants for over 20 years. Most of the installations are in European plants, however several are in use in the United States (e.g. Pilgrim Nuclear Power Station, Callaway Plant). The available operational history of the system-medium actuators show very good reliability and few instances of degraded performance.

Based on the simplicity of the design, the elimination of elastomeric seals, and available operational history, WCNOC believes that the valve performance will not degrade during the operational cycle and will perform its specified safety function.

(d) During performance of the surveillance test, steam generator pressure is the best available indication to the operators. This surveillance testing occurs during very low or no feedwater flow to the steam generators (in MODE 4 or MODE 3), such that the difference between the feedwater flow pressure and steam generator pressure would be insignificant.

2. In Section 4.8, page 85 of 133, item 3 of the input parameters, the fan cooler degradation in the GOTHIC analysis is assumed to be 20 percent as compared to 32 percent to 95 percent in the current licensing basis analysis. This has reduced the conservatism in the new analysis.

   In terms of table 4.8-1 of Attachment IV (page 86 of 133), what is the containment fan cooler heat removal rate that is used to determine that the fan cooler is operable, and how and how often is the surveillance made or performed? What is the assurance that the current heat removal rate would not go below the 20% degradation values listed in Table 4.8-1 between surveillance checks? Provide appropriate justification including vendor data as a basis for assuming 20 percent degradation of the fan cooler. What is the uncertainty in determining the actual heat removal rate of the fan coolers?

Response: Non-technical specification surveillance procedure STN PE-038, "Containment Cooler Performance Test," is used to verify the heat transfer capability of the containment fan coolers. This testing is performed based on WCNOC's response to Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment." This test is performed on one fan cooler, once per operating cycle (18 months). The acceptance criteria in STN PE-038 is the same as the value shown in Table 4.8-1 of Attachment I to ET 07-0004, 13387.5 Btu/sec (48,195,000 Btu/hr), corresponding to an inlet gas temperature of 277°F.

The results from the latest STN PE-038 surveillance demonstrated that containment fan cooler SGN01D was capable of removing 77,569,693 Btu/hr at accident conditions. The procedure also requires test uncertainty to be considered, and for this test, the uncertainty was calculated at 14,857,539 Btu/hr. Subtracting the test uncertainty leaves an adjusted heat transfer of 62,712,154 Btu/hr, which is above the minimum requirement of 48,195,000 Btu/hr.

Table 4.8-1 also lists a value of 15701.33 Btu/sec (56,524,788 Btu/hr) heat removal capability needed when allowing for 20% degradation. The results of the latest surveillance test exceed 56,524,788 Btu/hr, the minimum required value for 20% degradation.

Summary of Surveillance Test

Asset	Test Results (Btu/hr)	Test Uncertainty (Btu/hr)	Net Test Results (Btu/hr)	Minimum Requirement (Btu/hr)	20% Degradation (Btu/hr)
SGN01D	77,569,693	14,857,539	62,712,154	48,195,000	56,524,788

The latest containment fan cooler test shows that the cooler exceeds both the current minimum requirements and the proposed requirements assuming 20% degradation. The containment fan coolers have successfully passed prior heat transfer tests performed at an 18 month frequency. Therefore, it is expected that the fan coolers heat removal rate will not reduce to a rate below the 20% degradation values listed in Table 4.8-1 between surveillance tests.

Prior to 2005, the containment fan cooler test used an efficiency method to determine the coolers capability. Additionally, prior to 2005, test uncertainty was not calculated or considered in the test results. New software developed by a vendor was implemented in 2005, along with procedure changes to calculate and apply test uncertainty to the results. The following table provides the results of fan cooler testing performed since 2004.

Date	Test Results (Btu/hr)	Test Uncertainty (Btu/hr)	Net Test Results (Btu/hr)	Minimum Requirement (Btu/hr)	20% Degradation (Btu/hr)
10/19/05	80,677,086	15,355,056	65,322,030	48,195,000	56,524,788
05/02/07*	77,569,693	14,857,539	62,712,154	48,195,000	56,524,788

*During the test on 05/02/07 one tube bundle was isolated from service water flow.

It is important to note that the increase in the MSIV and MFIV stroke time from 5 seconds to 15 seconds results in additional feedwater being added to the faulted steam generator and causes additional mass and energy to be released to the containment prior to isolation. Consequently, the calculated peak containment pressure or temperature is expected to increase. Thus, the margin of the containment design pressure is reduced. In order to regain margin (> 10%), the degradation of the fan cooler performance was reduced uniformly by 20% from the vendor data, instead of the degradation values (i.e., 32% to 95%) assumed in the current Main Steam Line Break (MSLB) containment analyses.

The actual performance of the containment fan coolers under post-design basis accident conditions was predicted based on a computer program, developed by the vendor American Air Filter (AAF). The validity of the analysis and the computer program has been verified by performing a series of tests of the heat transfer capability of full-size finned coils utilized by several nuclear power plants. The comparison of test results with predicted heat transfer was documented in Topical Report AAF-TR-7101, "Design and Testing of Fan Cooler-Filter Systems for Nuclear Applications," dated February 20, 1972. This topical report was reviewed and found acceptable by the USAEC regulatory staff.

As presented in the topical report, agreement between predicted and measured performance appears quite good for conditions ranging up to 289°F at 68.3 psia. The benchmark comparison shows that the test results of heat transfer are 0% to 12% higher than the predicted values, whereas the test results and predicted values of condensate flow are within +10%/-3% to each other. This provides reasonable assurance that the actual heat removal rate of the fan coolers would be within +10%/-5% of the predicted values. In view of this vendor information, the 20% degradation assumption used for the containment analysis is judged to be conservative and justifiable.

3. In Section 3.2.3, page 10 of 133, first paragraph, there are the following statements: "The safety function of the MFRVs and MFRV bypass valves is to provide backup isolation of the main feedwater flow to the secondary side of the SGs [(steam generators)] following an HELB [(high energy line break)]. If the single active failure postulated for a secondary pipe break is the failure of a safety grade MFIV to close, then credit is taken for closing or isolating the non-safety grade MFRVs [(main feedwater regulating valves) or MFRV bypass valves. The MFRVs and MFRV bypass valves are highly reliable backups to the MFIVs."

   Provide closing time data and the failed position of the MFRVs and MFRV bypass valves with justification. Provide the results of an evaluation confirming that the mass and energy released inside the containment for the MFIV closure bounds the mass and energy release in case the MFIV fails to close and feedwater isolation is achieved by the MFRV and MFRV bypass valve.

For the MFRVs and MFRV bypass valves, provide closing time data and address the assurance that the valve closure time would not degrade below that needed for the accident analyses in the time interval between the times to test the valves, which is the proposed frequency of testing these valves within the inservice testing program.

Response: The addition of main feedwater to the faulted steam generator directly affects the mass inventory that may be released from the steamline break. A safety injection signal causes the main feedwater pumps to trip as well as the closure of the MFIVs, which terminates the pumped main feedwater flowrate. Another mechanism for main feedwater entering the faulted steam generator is due to flashing of the hot water that is in the unisolable section of the feedline, between the faulted steam generator and the closed isolation valve. The flashing occurs when the steam generator depressurizes below the feedwater saturation point and the decrease in density of the saturated water forces it from the feedline into the steam generator. Thus, the mass added to the faulted steam generator from both the pumped main feedwater and the feedline flashing may be larger if there is a failure of a MFIV.

The effects of any flashing of the feedwater trapped between the steam generator and the back-up isolation valve, assuming failure of a MFIV, is included in the analyses. The failure of the MFIV on the faulted loop results in additional fluid being added to the faulted steam generator. The quantity of the additional fluid to be released is based on the volume between the MFIV and the main feedwater regulating valve (MFRV) on the faulted loop.

Based on an examination of the integrated mass and energy released, the total MSLB blowdown mass and energy calculated for the MFIV failure case seems to be significantly higher than the case without the MFIV failure assumption. For instance, the calculated results for the MFIV failure case show total mass and energy releases of 611.1×10^3 lbm and 727.4×10^6 btu, respectively, resulting from a full double-ended rupture at 102% power level (i.e., Case 1). Whereas, the calculated results without a MFIV failure show total mass and energy releases of 585.5×10^3 lbm and 696.4×10^6 btu, respectively, for the Case 1 scenario. This represents an increase of approximately 4.5% for both mass and energy releases. The significant increase in the mass and energy released can be attributed to the increase in the unisolable feedline volume on the faulted loop as a result of the MFIV failure assumption. It is expected that the larger mass and energy releases will result in higher calculated containment pressures and temperatures.

The MFRVs and MFRV bypass valves are not currently included in the Inservice Testing Program and do not receive any periodic closure time testing. The closure time of the MFRVs and MFRV bypass valves was measured during the Feedwater System initial plant start-up testing. During Refueling Outage 15 (Fall 2006) the valve positioners were replaced and post modification testing was performed including measuring the closure time of the valves. The start-up tests results show MFRV bypass valve closure times of 9 – 12 seconds. The post modification tests show MFRV closure times of 3.5 – 3.8 seconds.

The probable sources of degraded closure time for these valves reside in the control components and valve packing. The control components, consisting of the I/P converter, positioner, etc., are industry standard types, and have records of good reliability and availability. The valve packing is of the live load type, and is adjusted during the operating cycle only if a leak develops. The following wording was proposed to be added to the TS SR 3.7.3.1 Bases,

> If it is necessary to adjust stem packing to stop packing leakage and if a required stroke test is not practical in the current plant MODE, it should be shown by analysis that the packing adjustment is within torque limits specified by the manufacturer for the existing configuration of the packing, and that the performance parameters of the valve are not adversely affected. A confirmatory test must be performed at the first available opportunity when plant conditions allow testing. Packing adjustments beyond the manufacturer's limits may not be performed without (1) an engineering analysis and (2) input from the manufacturer, unless tests can be performed after the adjustments.

Based on the set of control components with good records of reliability and the controls associated with valve packing adjustments, WCNOC believes that the MFRV and MFRV bypass valve performance will not degrade during the operational cycle and will perform its specified safety function.