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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION
DOCKET NOS. 50-445 AND 50-446
10CFR50.59 EVALUATION SUMMARY REPORT 0014 AND
COMMITMENT MATERIAL CHANGE EVALUATION
REPORT 0008

Dear Sir or Madam:

Please find attached the report required by 10CFR50.59(d)(2) for those activities which were completed or partially completed at Comanche Peak Units 1 and 2 between August 2, 2006, and February 1, 2008, and which were not reported to the NRC in a previous submittal. This report contains a brief description of the changes, tests and experiments implemented or performed pursuant to 10CFR50.59(c), including a summary of the evaluations for each. Items in this report are referenced by their 10CFR50.59 Evaluation Numbers. This report also includes certain activities completed or partially completed after February 1, 2008.

Luminant Generation Company LLC (Luminant Power) did not make commitment material changes which require reporting for Comanche Peak Units 1 and 2 per the recommendations of NEI document, "Guideline for Managing NRC Commitments," Revision 2. Therefore, no descriptions are provided for Commitment Material Change Evaluation Report 0008.

IE47
NRR

This communication contains no new licensing basis commitments regarding Comanche Peak Units 1 and 2.

Should you have any questions, please contact Mr. J. D. Seawright at (254) 897-0140.

Sincerely,

Luminant Generation Company LLC

Mike Blevins

By: Fred W. Madden

Fred W. Madden
Director, Oversight & Regulatory Affairs

Attachment

c - E. E. Collins, Region IV
B. K. Singal, NRR
Resident Inspectors, Comanche Peak

59EV-2003-002426-02-00
59EV-2003-002426-03-00
59EV-2005-000224-01-00
59EV-2006-000629-01-00
59EV-2006-003867-01-00
59EV-2007-000878-01-00
59EV-2007-001435-01-00
59EV-2008-000114-01-00

Evaluation Number: 59EV-2003-002426-02-00 **Unit 1**
Revision 0

Activity Description:
For Comanche Peak Unit 1 with the Delta-76 replacement steam generator design, an alternate methodology, described in WCAP-10325-P-A, was used to develop the LOCA mass and energy releases to be used in the evaluation of the containment design in lieu of the FSAR-described methodology in Section 6.2.

Summary of Evaluation:
The methodology used to develop the LOCA mass and energy (M&E) releases for use in the evaluation of the containment design for Comanche Peak Unit 1 with the Delta-76 replacement steam generators has been revised to be consistent with the methodology described in WCAP-10325-P-A. "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version." This methodology has been generically approved by the NRC for this application at Westinghouse pressurized water reactors. The methodology is used in accordance with the NRC's Safety Evaluation Report (SER), which is contained within WCAP-10325-P-A. As required by the SER, a plant-specific model was used to develop the M&E releases. All other constraints and limitations identified in the SER were met.

The use of the WCAP-10325-P-A methodology for developing LOCA M&E releases at Comanche Peak Unit 1 has been previously approved by the NRC for the intended application. As such, the proposed activity does not result in a departure from a method of evaluation described in the FSAR and used in establishing the design bases or in the safety analyses. Prior NRC approval is not required.

Evaluation Number: 59EV-2003-002426-03-00 **Unit 1**
Revision 0

Activity Description:
The MULTIFLEX 3.0 computer code was used in lieu of the FSAR-described methodology in Section 3.9N to calculate the hydraulic forces associated with reactor coolant system pipe breaks.

Summary of Evaluation:
In support of the Comanche Peak Unit 1 Replacement Steam Generator Program, the MULTIFLEX 3.0 computer code was used to calculate the hydraulic forces associated with the postulated pipe breaks in lieu of an earlier version of MULTFLEX described in FSAR Section 3.9N. The NRC has reviewed and approved the specific application of MULTIFLEX 3.0 for activities identical to the proposed activity at Comanche Peak. MULTIFLEX 3.0 has been extensively used (and referenced) in the development of hydraulic forces for input to vessel/internals/fuel LOCA dynamic analyses for two-, three-, and four-loop Westinghouse plants. Of specific applicability to Comanche Peak, the application of MULTIFLEX 3.0 in the development of hydraulic forces for the reactor vessel/internals/fuel components were identified in a June 3, 2004 Technical Change Request for the Indian Point Nuclear Generating Unit No. 3 (Docket No. 50-286), and reviewed and approved by the NRC as documented in a Safety Evaluation Report dated March 24, 2005. This SER was reviewed for limitations on the use of the code. In the SER, there were no limitations or conditions imposed on the use of MULTIFLEX 3.0 for these applications.

Because the application of MULTIFLEX 3.0 at Comanche Peak is identical to NRC-approved applications at similar plants, and because the code is being used within the limitations required by the NRC through the cited SER, the use of this code does not constitute a new or different method of evaluation that has not been approved by the NRC for the intended application. As such, the proposed activity does not result in a departure from a method of evaluation described in the FSAR and used in establishing the design bases or in the safety analyses. Prior NRC approval is not required.

Evaluation Number: 59EV-2005-000224-01-00 **Unit 1**
Revision 0

Activity Description:
The proposed activity is the use of the RELAP5Mod3.2 computer code in the development of time-dependent blowdown forces resulting from a postulated pipe rupture in high-energy piping other than the Reactor Coolant System main loop. The current Final Safety Analysis Report (FSAR) Section 3.6B.2.2.2 cites the use of an earlier version of RELAP5 which is no longer available.

Summary of Evaluation:
The proposed activity is the use of the RELAP5Mod3.2 computer code in the development of time-dependent blowdown forces resulting from a postulated pipe rupture in high-energy piping other than the Reactor Coolant System main loop. The current FSAR analyses are based on RELAP5Mod1.2. Despite its widespread use, the explicit approval by the NRC of RELAP5Mod3.2 for the development of blowdown forces was not identified for other plants. As such, the "conservative or essentially the same" argument described in the 50.59 Resource Manual (see also NEI-96-7, Revision 1) was used to address the proposed activity at Comanche Peak.

A formal calculation was developed to provide a comparison of results from a calculation using RELAP5Mod1.2 and a similar calculation using RELAP5Mod3.2. These two programs are compared by using the same RELAP input, i.e., the input for an original blowdown force calculation for the Comanche Peak main feedwater system was formatted for RELAP5Mod3.2. A limiting scenario from the original blowdown force calculation was selected for the comparison. The comparison of RELAP5Mod1.2 and RELAP5Mod3.2 analyses for the same transient case, showed peak pressures within 3.5% with similar timing of the peaks. The conclusion is that the results using the RELAP5Mod3.2 computer program are equivalent to results which would have been obtained with the RELAP5Mod1.2 computer program.

In summary, because the results with the RELAP5Mod3.2 computer code are essentially the same as results with the RELAP5Mod1.2 computer code described in the current FSAR, the proposed activity does not constitute a departure from a method of evaluation described in the UFSAR for the generation of fluid hydraulics forces resulting from pipe breaks in high-energy piping other than the RCS main loop piping. Therefore, prior NRC approval is not required.

Evaluation Number: 59EV-2006-000629-01-00 **Units 1& 2**
Revision 0

Activity Description:
Revise Technical Requirement (TR) 13.9.31, Decay Time, to allow irradiated fuel movement within the reactor vessel 75 hours after shutdown. The current requirement is 100 hours after shutdown. Additional design basis and licensing basis documents as discussed in the 10CFR50.59 Screen (59SC-2006-000629-01) are revised to support this Technical Requirements Manual (TRM) change.

This evaluation addresses the adverse impact of the increase in radiological consequences of a postulated Fuel Handling Accident (FHA) occurring at 75 hours after shutdown. The 10CFR50.59 Screen for the proposed activity addresses all other aspects of 10CFR50.59 review as these aspects were properly screened out.

Summary of Evaluation:
The proposed activity revises TRM TR 13.9.31 to allow irradiated fuel movement within the reactor vessel 75 hours after shutdown. The current requirement is 100 hours after shutdown. The 10CFR50.59 Screen of the proposed activity has determined that the radiological consequences of an FHA occurring at 75 hours are adverse relative to the current analysis and that further evaluation is required to determine if prior NRC approval is required. All other aspects of the proposed activity were properly screened out of the 10CFR50.59 review process.

The proposed activity does not modify the physical plant or change how plant equipment is controlled or operated but merely allows irradiated fuel movement to occur sooner than is currently allow by TRM TR 13.9.31. Additionally, all SSC design functions which support mitigation of the radiological consequences of an FHA or support assumptions of the FHA analysis are not adversely affected as described in the 10CFR50.59 Screen. Consequently, only 10CFR50.59(c)(2) criterion(iii) is potentially affected. All other criteria were determined to be unaffected by the proposed change.

The 10CFR50.59 Evaluation of the radiological consequences of a postulated FHA occurring at 75 hours determined the increases in accident doses for all locations and dose types were less than minimal and that significant margin to Regulatory and SRP dose limits is maintained. The limiting increase (3.0 rem) was found to be less than 10% of the difference between the current analysis and the 10CFR100 limit (28.1 rem) and the corresponding dose (22 rem) was found to be less than the SRP limit (75 rem). Consequently, the increase in radiological consequences of a postulated FHA at 75 hours is less than minimal and prior NRC approval is not required.

Evaluation Number: 59EV-2006-003867-01-00 **Unit 1**
Revision 0

Activity Description:
PCN-05 to MDA-316 Rev 0 is a procedural change to control bypassing of the Containment Crane Anti-Collision Control System (CCACCS). The change establishes the responsibilities and administrative controls for the Multi-Crane Coordinator in order to provide safe operation of the Polar Crane and Telescopic Jib Crane during 1RF12 in support of the Steam Generator Replacement Project.

The existing procedure addresses the use of bypass measures provided in the existing Containment Crane Anti-Collision Control System, CCACCS design and compensatory administrative controls to be used when the CACCS is degraded. The revision adds administrative controls which may be implemented to perform Steam Generator Replacement activities in 1RF12. These alternative administrative controls will maintain the minimal likelihood of unplanned crane interactions with the polar crane or other structures.

The alternative administrative controls being taken by this procedure change include the bypassing of some or all CCACCS features, the use of a second administratively control permissive bypass key, a heightened defense-in-depth by the use of a Multi-Crane Coordinator Oversight (from the top of Containment Building elevator - Elevation 933 or 28 feet ceiling), use of dedicated radio system for communications, color coding vest for crane operation identification, and special multi-crane operation training for the involved operators/riggers prior to operating multiple cranes in Containment.

Summary of Evaluation:
A Polar Crane and a Telescopic Jib Crane are provided inside Containment. These cranes are only used to handle loads during plant shutdown (e.g., MODES 5, 6, and defueled). Both cranes are provided with radio controls which allow remote operation (e.g. near the load). These cranes can operate in a Multi-Crane Operation mode - When two or more cranes are operating simultaneously in the same area, and there is a potential for crane-to-crane collision.

A Containment Anti-Collision Control System was installed on the Telescopic Jib Crane and the Polar Crane to reduce the likelihood of unplanned crane interactions with the polar crane or other structures. The function of the anti-collision sensor array, Programmable Logic Controller (PLC) and associated components is to stop the end of the Telescopic Jib Crane boom from approaching any structure within a set amount of distance and to keep the jib crane boom below the Polar Crane structure. In addition, an administratively controlled key, which is normally placed in the Polar Crane radio controller and allows full operation from that polar crane radio controller. In this mode, without the key in the Telescopic Jib Crane Controls, the PLC restricts the height of the Telescopic Jib Crane Boom. When the key is removed from the Polar Crane radio controller and placed in the Telescopic Jib Crane controls, the Telescopic Jib Crane boom height limit is bypassed and the polar crane bridge is locked out of being controlled by the radio controls. Operation of the trolley and hoists from the polar crane radio controls and full operation from the polar crane cab is still available under the administrative controls for multi-crane operation. These administrative controls include a multi-crane coordinator located in a position of good visibility to observe both cranes.

As indicated and described above, the anti-collision system provides an automated control to minimize the likelihood of unplanned crane interactions with the polar crane or other structures. However, it does not prevent such interaction from occurring due to hardware or software failures. Human actions are still required as a measure to also reduce such crane/load interactions. Although the design and administrative controls cannot absolutely preclude any interactions, it was concluded that catastrophic failures that could result in multiple missiles have been reduced below the level considered credible in the licensing basis (i.e. comparable to other equipment).

The described control measures in the Final Safety Analysis Report (FSAR) are one means of utilizing the existing installations to provide for the use of the automated controls system. However, it is not the only way of limiting crane/load interactions. The FSAR also recognizes that simultaneous full operations of the jib crane and polar crane may be required. Thus, the FSAR describes the normal way of controlling the full operation of both cranes by the use of the control key in the jib crane control and operation of the polar crane from the cab with the additional administrative controls including a multi-crane coordinator located in a position of good visibility to observe both cranes.

The proposed change in the procedure has resulted in another way of providing full operation of both cranes by including alternative administrative controls for multi-crane operations and, when approved, the potential use of 2 control keys (one in the jib crane control and one in the radio controller for the polar crane).

The actions being taken by procedural administrative controls include the bypassing of the CCACCS features, use of a second administratively control permissive bypass key and a heightened defense-in-depth by the use of a Multi-Crane Coordinator Oversight (from the top of Containment Building elevator - Elevation 933 or 28 feet ceiling), dedicated radio system for communications, color coding vest for crane operation identification, and special multi-crane operation training for the involved operators/riggers prior to operating multiple cranes in Containment. These controls are judged to be equivalent or better than the controls provided by the CCACCS.

Based on the proposed alternative administrative actions described in the change to procedure MDA-316, it is concluded that effective and equivalent alternate administrative controls have been provided to the Multi-Crane Coordinator for heightened defense-in-depth measures to reduce and minimize crane interactions. It is concluded that the activity does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an structure, system, or component important to safety previously evaluated in the FSAR

Therefore, the change may be implemented under 10CFR50.59.

Evaluation Number: 59EV-2007-000878-01-00 **Units 1 & 2**
Revision 0

Activity Description:
Design Modification Authorization DMA-2002-000100-01 and -02 modified the low pressure turbine to replace the original Unit 1 ten-disc (five discs per flow, 44-inch last-stage blades) Low Pressure Turbine rotors with an eight-disc (four discs per flow, 46-inch last-stage blades) advanced design turbine rotor and associated components to provide efficiency improvements and reduced maintenance requirements which result in cost savings. Also, the first two discs per flow of the original Low Pressure Turbine rotors were combined into a single disc for the upgrade Low Pressure Turbine rotors. The efficiency improvements convert more thermal energy into mechanical energy, rather than being lost to friction, steam velocity changes and blade bypass. The upgrade rotor discs are made from 3.5% NiCrMoV steel material. This design improvement also increased disc inspection interval up to 100,000 equivalent operating hours.

Design Modification Authorizations DMA-2004-000773-01 and -02 was a modification to the turbine controls to upgrade the existing Turbine-Generator Protection Systems controls to a Siemens TXP system digital .Systems modified include the Turbine Trip System (TTS), Extended Turbine Protection (ETP), Electronic Generator Protection (EGP), Automatic Turbine Tester (ATT), and Vibration Expansion Measurement (VEM). Also included will be certain Turbine Trips associated with the Thyristor Voltage Regulator (TVR) and Generator Primary Water (PR) systems.

The change in methodology used for the calculation of the risk for damage to safety related SSCs for a postulated turbine generated missile was evaluated separately under 59EV-2002-000100-01 as required. Digital controls were evaluated separately under 59EV-2004-000773-01-03. However, the increase in probability of occurrence of turbine missile per turbine year (P1) and the quantitative change in the probability of an overspeed trip failure (P10) is evaluated herein.

Summary of Evaluation:
Design Modification Authorization 2002-000100-01 and -02 modified the low pressure turbine to replace the original Unit 1 Low Pressure Turbine rotors and increased disc inspection interval up to 100,000 equivalent operating hours. Since the vendor (Siemens) methodology does not compute P1 per year, the NRC acceptance limit is adjusted for comparison. P10 of 1E-04 per year for 100,000 hours is the equivalent of 11.42 E-04 at 100,000 hours. Reports TR-04108 (Proprietary) and TP-04124 (Non-Proprietary) document NRC approval of this adjustment to the NRC criteria. After the modification, P1 was 3.43E-5 at 100,000 hours (i.e., inspection interval). Since the P1 value is less than the NRC acceptance value (11.42 E-04 at 100,000 hours), the modification and change to the inspection interval satisfy 10CFR50.59 Criterion (i) and did not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR..

The value of P1 was again changed by resulting from an upgrade to the turbine controls from the existing Turbine-Generator Protection Systems controls to a Siemens TXP digital system.

Therefore, after both modifications, P1 resulted in 3.69E-05 at the 100,000 hours inspection interval. This value results in an accident frequency below 1E-07 per year for likelihood of an accident which meets the applicable plant specific accident threshold. Therefore, both turbine modifications satisfy 10CFR50.59 criterion (i) and did not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

The design modification to upgrade to the turbine controls from the existing Turbine-Generator Protection Systems controls to a Siemens TXP digital system also affected the probability of an overspeed trip failure (P10).

Based on "Comparison MTBF Evaluation Comanche Peak ST Protection and Trip system and Engineering Report No. ER-504 Probability of Turbine Missiles", Version: 2 dated October 24, 2004, the new total system failure probability is 3.10 E-06 compared to the pre-modification value of 3.23 E-06. This was a reduction in the probability of an overspeed trip failure.

This evaluation concluded the controls modification did not result in more than a minimal increase in likelihood of a malfunction of the overspeed trip system.

Therefore, both modifications met the applicable 10CFR50.59 criteria and NRC approval was not required.

Evaluation Number: 59EV-2007-001435-01-00 **Units 1& 2**
Revision 0

Activity Description:
Per DBD-ME-255, the maximum allowed letdown in Mode 1 and 2 is via two letdown orifices (i.e. the 45 gpm orifice and one of two 75 gpm orifices). The system design assures the letdown rate is less than or equal to 140 gpm. This letdown rate and configuration are key parameters for the evaluation of letdown line breaks outside containment. In Mode 4, all three orifices may be opened. In addition, once the Residual Heat Removal System (RHRS) has been started, letdown purification flow is provided downstream of the RHR heat exchanger. There is no change to the above DBD requirements in Mode 1 and 2.

The change is to allow an increase in maximum CVCS letdown flow rates from 140 gpm to 195 gpm to allow a more rapid clean up of the RCS during and after crud burst in Modes 3 (below 500 degrees F), 4 & 5. In order to increase flow above 140 gpm in Mode 3 < 500 F, the third letdown orifice must be opened. This results in an increase in mass & energy and radiological consequences in the event of a CVCS letdown line break outside containment. The increased process flow rates also exceed the original design of the plant. The impact on design and nuclear safety have been evaluated and found acceptable based on flow rate limitations and the limited duration when design flow would be exceeded.

DBD precautions and limitations required by this change are:
Higher letdown flow rates are acceptable under the following limitations:
* Letdown flow is limited to 140 gpm through the 45 gpm orifice and one of the 75 gpm orifices when RCS temp is > 500 degrees.
* The third orifice may not be opened when RCS temp is > 500 degrees.
* Letdown flow is limited to 170 gpm (when RCS temp is < 500 degrees) with 1 demineralizer in service.
* Letdown flow is limited to 195 gpm (when RCS temp is < 500 degrees) with 2 demineralizers in service.

Summary of Evaluation:
These changes to the DBD and to the procedures do not change the failure mode of any system, structure, or component. The new conditions were evaluated to be within the original design pressures and temperatures. The higher flow rates were evaluated and found either to be within design or acceptable based on the minor amount of flow over design and the limited time flow would be above design. Any impact on frequency of malfunctions or accidents would be less than minimal. No new accidents or malfunctions are created by this change.

The change does not affect any design basis limit for a fission product barrier.

The pertinent accident previously analyzed in the FSAR is a letdown line break outside containment. This accident was evaluated in Section 15.4.6 as a Condition III event for radiological consequences and in Section 3.6B for high energy line break (HELB) dynamic and environmental consequences.

The increase in HELB environments were minor and did not significantly change equipment qualification parameters.

The methods of evaluation used for this change are as described in the FSAR. The radiological consequences were evaluated using the same methodology, the same plant design inputs, and the same assumption of failed fuel as the current licensing basis in the FSAR. Only the direct effect of the change (increased mass & energy) was used for the evaluation of the consequences.

In order to increase flow above 140 gpm in Mode 3 < 500 F, the third letdown orifice must be opened. This results in an increase in radiological consequences in the event of a CVCS letdown line break outside containment. In Mode 3, below 500 degrees F, with three letdown orifices open (in lieu of two), the dose at the EAB is conservatively estimated to be increased by less than 57%. This results in 8.9 Rem to the thyroid and 0.061 Rem to the whole body. The LPZ doses are conservatively estimated to be 1.33 Rem to the thyroid and 0.0091 Rem to the whole body. As expected, the radiological consequences resulting from the failure of the 3 inch CVCS letdown line in Mode 3 do not exceed a small fraction of the dose values set forth in 10CFR100. In addition, the increase due to the opening of the third orifice is less than 10 percent of the difference between the current calculated dose value and the regulatory guideline value (10 CFR 100).

The new estimated control room doses are:

Thyroid - 1.0 Rem
Whole Body - 0.033 Rem
Beta Skin - 0.85 Rem

The increase in control room dose due to the opening of the third orifice is less than 10 percent of the difference between the current calculated dose value and the regulatory guideline value (GDC-19).

Therefore, the change does not result in more than a minimal increase in the consequences of a letdown line break.

In summary, this change meets the criteria in 10CFR50.59 and may be implemented without obtaining a license amendment.

Evaluation Number: 59EV-2008-000114-01-00 **Unit 2**
Revision 0

Activity Description:
Evaluate bypassing the anti-collision circuitry for 2RF10.

PCN-7 to MDA-316 Rev 0 is a procedural change to control bypassing of the Containment Crane Anti-Collision Control System (CCACCS). The change establishes the responsibilities and administrative controls for the Multi-Crane Coordinator in order to provide safe operation of the Polar Crane and Telescopic Jib Crane during 2RF10 in support of the spring outage.

The existing procedure addresses the use of bypass measures provided in the existing Containment Crane Anti-Collision Control System, CCACCS design and compensatory administrative controls to be used when the CACCS is degraded. The revision adds administrative controls which may be implemented during refueling outage 2RF10. These alternative administrative controls will maintain the minimal likelihood of unplanned crane interactions with the polar crane or other structures.

The alternative administrative controls being taken by this procedure change include the bypassing of some or all CCACCS features, the use of a second administratively control permissive bypass key, a heightened defense-in-depth by the use of a Multi-Crane Coordinator Oversight (from the top of Containment Building elevator - Elevation 933 or 28 feet ceiling), use of dedicated radio system for communications, color coding vest for crane operation identification, and special multi-crane operation training for the involved operators/riggers prior to operating multiple cranes in Containment.

Summary of Evaluation:
A Polar Crane and a Telescopic Jib Crane are provided inside Containment. These cranes are only used to handle loads during plant shutdown (e.g., MODES 5, 6, and defueled). Both cranes are provided with radio controls which allow remote operation (e.g. near the load). These cranes can operate in a Multi-Crane Operation mode - i.e. when the two cranes are operating simultaneously in the same area, and there is a potential for crane-to-crane collision.

The existing system is described as follows. A Containment Anti-Collision Control System was installed on the Telescopic Jib Crane and the Polar Crane to reduce the likelihood of unplanned crane interactions with the polar crane or other structures. The function of the anti-collision sensor array, Programmable Logic Controller (PLC) and associated components is to stop the end of the Telescopic Jib Crane boom from approaching any structure within a set amount of distance and to keep the jib crane boom below the Polar Crane structure. In addition, an administratively controlled key, which is normally placed in the Polar Crane radio controller and allows full operation from that polar crane radio controller. In this mode, without the key in the Telescopic Jib Crane Controls, the PLC restricts the height of the Telescopic Jib Crane Boom. When the key is removed from the Polar Crane radio controller and placed in the Telescopic Jib Crane controls, the Telescopic Jib Crane boom height limit is bypassed and the polar crane bridge is locked out of being controlled by the radio controls. Operation of the trolley and hoists from the polar crane radio controls and full operation from the polar crane cab is still available under the administrative controls for multi-crane operation. These administrative controls include a multi-crane coordinator located in a position of good visibility to observe both cranes.

As indicated and described above, the anti-collision system provides an automated control to minimize the likelihood of unplanned crane interactions with the polar crane or other structures. However, it does not prevent such interaction from occurring due to hardware or software failures. Human actions are still required as a measure to also reduce such crane/load interactions. Although the design and administrative controls cannot absolutely preclude any interactions, it was concluded that catastrophic failures that could result in multiple missiles have been reduced below the level considered credible in the licensing basis (i.e. comparable to other equipment).

The described control measures in the Final Safety Analysis Report (FSAR) are one means of utilizing the existing installations to provide for the use of the automated controls system. However, it is not the only way of limiting crane/load interactions. The FSAR also recognizes that simultaneous full operations of the jib crane and polar crane may be required. Thus, the FSAR describes the normal way of controlling the full operation of both cranes by the use of the control key in the jib crane control and operation of the polar crane from the cab with the additional administrative controls including a multi-crane coordinator located in a position of good visibility to observe both cranes.

The proposed change in the procedure has resulted in another way of providing full operation of both cranes by including alternative administrative controls for multi-crane operations and, when approved, the potential use of 2 control keys (one in the jib crane control and one in the radio controller for the polar crane).

The actions being taken by procedural administrative controls include the bypassing of the CCACCS features, use of a second administratively control permissive bypass key and a heightened defense-in-depth by the use of a Multi-Crane Coordinator Oversight (from the top of Containment Building elevator - Elevation 933 or 28 feet ceiling), dedicated radio system for communications, color coding vest for crane operation identification, and special multi-crane operation training for the involved operators/riggers prior to operating multiple cranes in Containment. These controls are judged to be equivalent or better than the controls provided by the CCACCS.

Based on the proposed alternative administrative actions described in the change to procedure MDA-316, it is concluded that effective and equivalent alternate administrative controls have been provided to the Multi-Crane Coordinator for heightened defense-in-depth measures to reduce and minimize crane interactions. It is concluded that the activity does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the FSAR

Therefore, the change may be implemented under 10CFR50.59.