La Salle 2
1Q/2011 Plant Inspection Findings

Initiating Events

Significance: G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Perform Adequate Evaluation for Reactor Building Crane Upgrade
During an inspection of pre-operational testing activities of an independent spent fuel storage installation (ISFSI) at the LaSalle County Station, the inspectors identified a finding of very low safety significance with an associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to perform adequate evaluations to upgrade the single failure proof crane. Specifically, the inspectors identified five examples where the licensee failed to perform adequate evaluations in accordance with American Society of Mechanical Engineers (ASME) NOG-1-2004, "Rules for Construction of Overhead and Gantry Cranes (Top Running and Bridge, Multiple Girder)," requirements. The Reactor Building crane is designed to Seismic Category I requirements and the licensee used compliance with ASME NOG-1-2004 as the design basis for their crane upgrade to a single failure proof crane. The inspectors determined that the failure to perform adequate evaluations was contrary to ASME NOG-1-2004 requirements and was a performance deficiency. The licensee documented the conditions in Issue Report (IR) 957014, IR 1093028, and IR 1098435 and initiated actions for calculation revisions and field modifications.
The finding was of more than minor significance because the failure to perform adequate evaluations affected the licensee's ability to provide reasonable assurance that loads would not be dropped during critical lifts. The inspectors evaluated the finding using Inspection Manual Chapter (IMC) 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," and based on a "No" answer to all of the questions in the Initiating Events column of Table 4a, determined the finding to be of very low safety-significance (Green). This finding has a cross-cutting aspect in the area of Human Performance, Work Practices because the licensee did not ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported. (IMC 0310, H.4(c)) (Section 4OA5)

Inspection Report# : 2010005 *(pdf)*

Mitigating Systems

Significance: G Mar 31, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Post Protected Pathway Signs for a Red Risk Path System
A finding of very low safety significance and associated NCV of 10 CFR 50.65(a)(4), Maintenance Rule, was identified by the inspectors for the licensee's failure to implement all necessary prescribed risk management actions during a Unit 2 RCIC system maintenance window. Specifically, the licensee failed to post protected equipment signs for Unit 2 systems whose unavailability would have taken the unit into a Red risk condition.
Description: On January 11, 2011, during a protected pathway walkdown for the Unit 2 RCIC work window, the inspectors identified that the Division III 2B Diesel Generator (DG), the HPCS dedicated diesel, and its associated room ventilation were not labeled with the appropriate protected equipment signs in the field. Further reviews revealed that this equipment was also not listed in the Protected Equipment Log (OP LA 101 111 1002, Att. B), kept by operations staff in the control room. These two systems were, however, listed in the Paragon online risk assessment program as Red risk equipment that should be protected according to the station's Protected Equipment Program procedure, OP AA 108 117, during the RCIC work window.

The inspectors determined that the licensee's failure to post protected equipment signs for Unit 2 systems whose unavailability would have taken the unit into a Red risk condition was contrary to the station's Protected Equipment Program procedure, OP AA 108 117, and is a performance deficiency. The inspectors determined that this finding is more than minor because the licensee failed to implement prescribed compensatory measures of posting signs and barricades to protect the HPCS equipment during the RCIC work window. Noteworthy was that while the licensee probabilistic risk assessment staff did identify the HPCS system and components as needing to be protected during the work planning stage, plant operators later used information from a separate (older) risk evaluation that did not include the HPCS equipment as needing to be protected. Hence, workers started to perform RCIC work without the HPCS equipment being protected until it was identified by the inspectors
Inspection Report# : 2011002 *(pdf)*

Significance: G Jan 03, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
Supporting Structure for Standby Liquid Control System Test Tank Non-Functional During Postulated Design Basis Earthquake (DBE).
The team identified a finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to have an adequate calculation to demonstrate the seismic qualification of the standby liquid control (SBLC) system test tanks. Specifically, the licensee could not ensure that the Units 1 and 2 SBLC test tanks, if filled with water, would not collapse and damage nearby safety-related components during a design basis event. The licensee entered this finding into their corrective action program and drained the water from the SBLC test tanks to restore seismic qualification.

The team determined that this finding was more than minor because it was associated with the Mitigating Systems cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability of the SBLC system to respond to initiating events to prevent undesirable consequences (i.e., core damage). This finding was determined to be of very low safety significance (Green) utilizing the Risk-Assessment Standardization Project Handbook based on the frequency of seismic events. The finding did not have a cross-cutting aspect because it was not reflective of current performance. (Section 1R21.3.b.(1))
Inspection Report# : 2010006 *(pdf)*

Significance: G Jan 03, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
EDG Usable Fuel and RHR Pump NPSH Calculations Failed to Consider Appropriate EDG Frequency Variations
The team identified a finding of very low safety significance and an associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control" for the licensee's failure to account for allowable frequency variations on the emergency diesel generators (EDG) in the diesel fuel oil consumption and residual heat removal (RHR) pump net positive suction head (NPSH) calculations. Specifically, the team noted the calculations assumed a frequency of 60 Hz whereas the Technical Specifications (TS) allowed steady state operation at a frequency of up to 61.2 Hz. The licensee entered this finding into their corrective action program and implemented a standing order and procedural limitations to ensure an adequate supply of fuel was available.

The team determined that this finding was more than minor because it was associated with the Mitigating Systems cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of the EDGs to respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, operating the EDGs at a frequency of 61.2 Hz would result in higher fuel consumption and reduced RHR pump NPSH margins. The finding is of very low safety significance (Green) because it did not result in a loss of operability. This finding had a cross-cutting aspect in the area of problem identification and resolution, operating experience because the licensee did not properly evaluate relevant operating experience. (P.2(a)) (Section 1R21.3.b.(2))
Inspection Report# : 2010006 *(pdf)*

Significance: Jan 03, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
Insufficient Design Bases for Degraded Voltage Time Delay and LOV Relay Settings
The team identified a finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” involving the licensee’s failure to have appropriate analyses for the loss of voltage relay setpoints and the second level undervoltage [degraded voltage] relay timer settings. Specifically, licensee’s analysis and technical basis for the auxiliary power system (AP) second level undervoltage relay time delay settings failed to demonstrate the ability of the permanently connected safety-related loads to continue to operate during the 5.5 minutes relay time delay without sustaining damage during a worst case, non-accident degraded voltage condition (when voltage was still above the setpoint of the loss of voltage relay setpoint). The licensee entered this finding into their corrective action program to verify the adequacy of the degraded voltage relay setpoint and time delay design.

The team determined that this finding was more than minor because the finding was associated with the Mitigating Systems Cornerstone attribute of Design Control, and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, there was reasonable doubt as to whether the permanently connected safety-related loads would remain operable during a worst case, non-accident degraded voltage condition for the duration of the time delay chosen for the degraded voltage relay. The finding was of very low safety significance (Green) since the existing settings for the inverse time relay currently being used for the loss of voltage relay would limit the duration of degraded voltage below 75 percent to only a few seconds. This finding had a cross-cutting aspect in the area of problem identification and resolution because similar concerns raised at the Byron Nuclear Station, during the 2009 CDBI, were not promptly evaluated and correctly dispositioned at LaSalle. [P1(c)] (Section 1R21.3.b.(3))
Inspection Report# : 2010006 *(pdf)*

Significance: Jan 03, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
Fast Transfer Scheme
The team identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” involving the licensee’s failure to analyze the capability of the electrical system to transfer safety related 4160V buses as described in the Updated Final Safety Analysis Report (UFSAR). The licensee entered this finding into their corrective action program and issued a standing order restricting alignment of safety buses to the unit auxiliary transformer (UAT) pending resolution of this issue.

The team determined that this finding was more than minor because it was associated with the Mitigating Systems cornerstone attribute of Design Control, and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was of very low safety significance (Green) since the safety buses had not been aligned to the UAT, the team determined the finding design deficiency did not result in loss of operability or functionality. The team did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. (Section 1R21.3.b.(4))
Inspection Report# : 2010006 *(pdf)*

Significance: Dec 31, 2010
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Failure to follow the performance centered monitoring process specified in procedure MA-AA-716-210
On September 25, 2010, the supply and exhaust fans for the Unit 2 Division 3 switchgear room ventilation system (VD) were unexpectedly found tripped. Division 3 switchgear supports the high pressure core spray (HPCS) system. Following this discovery, all Unit 2 Division 3 equipment was declared inoperable and unavailable. As HPCS is a single train system, this failure resulted in a complete loss of system function, requiring the licensee to make an eight hour notification to the NRC under 10 CFR 50.72(b)(3)(v)(D) and subsequent LER under 50.73(a)(2)(v)(D). The relay was replaced and tested satisfactorily. The cause of the relay failure was subsequently determined to be age-related

degradation.
The inspectors reviewed the event described in LER 05000374/2010-01-00 for accuracy and potential violations. In addition, as part of the assessment, the inspectors evaluated the extent of condition review and the adequacy of the corrective actions performed by the licensee.
Inspection Report# : 2010005 *(pdf)*

Significance: G Jun 30, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to develop and implement an adequate surveillance test procedure to accurately assess the as found trip setpoint for the pressure switches associated with the main steam line low pressure isol
This NCV is for licensee failure to develop and implement an adequate surveillance test procedure to accurately assess the as found trip setpoint for the pressure switches associated with the main steam line low pressure isolation function and various other safety-related functions.
During a followup review of Task Interface Agreement 2009 006 “Unacceptable Preconditioning of Safety Related Pressure Switches During Required Surveillance Testing at Montcello” issued by the NRC in September 2009, the inspectors identified that the licensee’s surveillance testing procedures established a methodology which tested various safety related pressure switches in a manner which was deemed unacceptable preconditioning by the NRC. In particular the inspectors noted that during the LIS MS 101A(201A) procedure, “Unit 1(2) Main Steam Line Low Pressure MSIV Isolation Calibration in Run Mode” Revision 5, the pressure switches in question were initially subject to main steam pressure. In accordance with the surveillance procedure, the inspectors noted that the basic testing methodology associated with these pressure switches was as follows: 1) isolate the pressure switch to be tested; 2) uncap the test connection; 3) connect the test equipment to the test connection; 4) increase the pressure until the pressure switch resets and record the reset test data; 5) bleed off the pressure until the pressure switch trips and record the as found trip setpoint; 6) remove the test equipment and restore the pressure switch to operation. This testing methodology caused the pressure switch and associated contacts to change state when the system pressure was relieved in Step 2; again when pressure was applied to reset the pressure switch in Step 4; then a third time when the pressure was bled off to obtain the as found trip setpoint in Step 5. This testing methodology subjected the pressure switch to a maximum pressure differential (operating pressure to atmospheric) and fully cycled the pressure switch prior to obtaining the as found trip setpoint data. This particular surveillance was most recently performed on unit 1 MSIV pressure switches on April 16, 2010 and on unit 2 MSIV pressure switches on June 11, 2010. The inspectors review also identified that no engineering justification had been performed by the licensee to show that testing of these pressure switches in the above manner did not impact the accuracy and reliability of the safety related pressure switches.
The inspectors noted that the existing licensee pressure switch testing methodology ensured operability of the pressure switches subsequent to the performance of the applicable surveillance test, since the required as left pressure switch setpoint was adjusted (if required) prior to the completion of the surveillance. The inspectors determined that the existing testing methodology potentially masks existing conditions; such as sticking contacts, mechanical binding, and setpoint drift; and could mask existing operability concerns because the pressure switch is fully cycled prior to obtaining the as found trip setpoint data.
Inspection Manual Chapter (IMC) 9900 states, in part, that unacceptable preconditioning is defined as the alteration, variation, manipulation or adjustment of the physical condition of a SSC before or during TS surveillance or American Society of Mechanical Engineers code testing that will alter one or more of SSCs operational parameters, which results in acceptable test results. Such changes could mask the actual as found condition of the SSC and possibly result in an inability to verify the operability of the SSC. In addition, unacceptable preconditioning could make it difficult to determine whether the SSC would perform its intended function during an event in which the SSC might be needed. Therefore, the inspectors concluded that since the licensee had not performed an evaluation which justified that the preconditioning of the pressure switches was acceptable, the licensee’s surveillance testing methodology which cycles a pressure switch prior obtaining as found trip setpoint data constituted unacceptable preconditioning of the pressure switch.
Further investigation by the inspectors revealed that an additional 36 pressure switches in Units 1 and 2, which are relied upon to initiate TS related protective functions in the areas of emergency core cooling system low pressure injection permissive, TCV fast closure, main condenser low vacuum scram, reactor core isolation cooling (RCIC) steam low pressure isolation, and reactor high pressure shutdown cooling isolation were tested in a manner similar to that described above with no engineering justification.
Analysis: The inspectors determined that the failure to develop and implement an adequate surveillance test procedure

to accurately assess the as found trip setpoint for the pressure switches associated with the main steam line low pressure isolation function and other safety related functions constituted a performance deficiency warranting significance evaluation in accordance with IMC 0612, Appendix B, “Issue Disposition Screening.” The inspectors determined that the performance deficiency was more than minor and a finding because it impacted the Reactor Safety Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences and affected the cornerstone attribute of Equipment Performance. The inspectors did not identify any cross cutting aspects associated with this finding.
The inspectors evaluated the finding using IMC 0609, Appendix A, Attachment 1, “Significance Determination of Reactor Inspection Findings for At Power Situations,” using the Phase 1 Worksheet for the Initiating Events Cornerstone. Since the inspectors answered all of the Exhibit 1, Table 4a Mitigating Systems questions no, the inspectors concluded that the finding was of very low safety significance.
Enforcement: Title 10 CFR, Part 50, Appendix B, Criterion V “Instructions, Procedures and Drawings”, states, in part, that activities affecting quality shall be prescribed by documented instructions of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, most recently on June 11, 2010, the licensee failed to prescribe a documented instruction that was appropriate to the circumstances for the testing of the pressure switches for the Main Steam Low Pressure Group I Isolation, an activity affecting quality. Specifically, Procedure LIS MS 201A incorporated a testing methodology that inappropriately manipulated the pressure switches prior to obtaining as found data, thus resulting in unacceptable pre conditioning. Because this violation was of very low safety significance and was entered into the licensee’s CAP (Issue Report (IR) 988976), it is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy.

Inspection Report# : 2010003 *(pdf)*

Barrier Integrity

Significance: G Mar 31, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Follow Plant Barrier Control Process Caused Secondary Containment to Become Inoperable
A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was identified by the inspectors for the licensee’s failure to follow steps 3.6 and 3.7 of procedure CC AA 201, Revision 8, entitled “Plant Barrier Control Program.” Specifically, evidence has shown that two adjacent airlock doors were opened simultaneously for a period of time sufficient to cause reactor building air pressure to surpass the TS allowed value for operability.
Description: Through a review of operator logs, the inspectors noted that on March 3, 2011, at 9:28 a.m., while Unit 1 was in Mode 1 and Unit 2 was in Refueling Mode in the process of loading fuel into the reactor, reactor building to outside differential pressure was observed to be in excess of 0.25 inches of water column—the maximum value allowed by TS Surveillance Requirement 3.6.4.1.1. As a result, TS Limiting Condition for Operability (LCO) 3.6.4.1 Condition A was entered for both units, since the reactor building is a shared structure, and core alterations were immediately halted. The condition was immediately evaluated by the licensee and entered into the CAP as AR 1182255, entitled “Loss of Secondary Containment During Fuel Movement.” Shortly thereafter, secondary containment pressure returned to the acceptable range, the licensee declared secondary containment operable at 9:42 a.m., and fuel moves recommenced by 10:00 a.m.
This AR documented the licensee’s decision making process for troubleshooting the pressure anomaly. The licensee identified that there were two potential causes for the loss of differential pressure in the reactor building: a malfunctioning ventilation system, or a breach of the ventilation envelope of the reactor building. Instrument maintenance personnel were immediately dispatched to the operating reactor building ventilation panel and found no evidence that the ventilation system was malfunctioning. Additionally, due to previous operating experience at LaSalle, which has shown that the simultaneous opening of two reactor building airlock doors causes similar effects on secondary containment pressure, the licensee promptly dispatched operations personnel to observe various interlocked door locations to check for openings. Although no doors were reportedly found open at the time, the licensee nevertheless concluded that since the reactor building ventilation system apparently functioned as designed, the loss of operability of the secondary containment was due to the opening of two interlock doors.

The inspectors conducted interviews with the site engineering staff responsible for secondary containment ventilation as well as for the design and operation of the door interlock systems. Through these discussions, and the review of ventilation system flow output traces for the period of time in question, the inspectors noted that the preponderance of evidence pointed towards a cause of two interlocked airlock doors being held open by plant workers. This opening would have represented an unmonitored pathway to the environment from the reactor building in the case of a fuel handling accident on Unit 2 or a design basis accident on the operating Unit 1. Additionally, it was explained to the inspectors by the engineering staff that the open pathway existed for approximately 15 minutes before being closed. The period of time that the building pressure actually exceeded the TS limit was approximately 3 minutes of that 15 minute window.

The associated Traditional Enforcement Item for failure to make the associated report t oNRC is item 2011-002-03.
Inspection Report# : 2011002 *(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

Miscellaneous

Significance: SL-IV Mar 31, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Make a Required 10 CFR 50.72 report for an Inoperable Secondary Containment
A Severity Level IV NCV of 10 CFR 50.72 (b)(3)(v) was identified by the inspectors for the failure of the licensee to make an eight hour report to the NRC for a condition that, at the time of discovery, could have prevented secondary containment from fulfilling its safety function. Specifically, when two adjacent reactor building airlock doors were opened concurrently, the ability of the system to perform its specified safety function could no longer be assured.

Description: On March 3, 2011, reactor building to outside differential pressure was observed to be in excess of 0.25 inches of water column—the maximum value allowed by TS Surveillance Requirement 3.6.4.1.1. As a result, TS LCO 3.6.4.1, Condition A, was entered for both units.
The inspectors agreed with the licensee's conclusion that since the ventilation system operated as expected, the condition could have only been caused by the concurrent opening of two adjacent reactor building airlock doors by plant workers. The open, unmonitored, unfiltered pathway that was created essentially defeated the safety function of the secondary containment building by allowing a bypass pathway for radioactive effluent to escape in the event of a design basis accident on the operating unit or a fuel handling accident on the refueling unit. See section 1R15.(1)

above for more details.
LaSalle's TSs define OPERABLE OPERABILITY by stating "A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s)…" Therefore, by applying LaSalle's definition of OPERABILITY, when a system is declared INOPERABLE it is incapable of performing its specified safety function(s). In the case of secondary containment, when it was declared inoperable for failing to maintain > 0.25 inches of vacuum, the secondary containment, by definition, had been deemed incapable of performing its specified safety function.
Further, in the review of this issue, the inspectors analyzed the challenge to the configuration of the reactor building ventilation envelope through a snapshot in time approach. Specifically, at the moment that both doors were opened simultaneously and not immediately shut, the operability of the secondary containment would have been immediately brought into question, regardless of the building differential pressure being above the TS allowed value at the time. The reason is due to TS SR 3.0.1, which states, in part, that "Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO." In the snapshot in time where an open pathway existed in the secondary containment boundary, SR 3.6.4.1.3, which requires that the secondary containment be drawn down to > 0.25 inch of vacuum water gauge in < 900 seconds using one standby gas treatment subsystem, may not have been able to have been satisfied; therefore, in accordance with SR 3.0.1, the LCO may not have been met. This is further evidence that the operability and ability to perform its safety function were not ensured when two redundant airlock doors were opened simultaneously.
In accordance with NUREG 1022, Event Reporting Guidelines – 10 CFR 50.72 and 50.73, Revision 2, an event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to control the release of radioactive material, the condition identified on March 3 should have been reported within eight hours to the NRC under 10 CFR 50.72(b)(3)(v). The licensee did not do so within the required timeframe.
Analysis: The inspectors determined that the failure to make a required eight hour report to the NRC for the loss of the secondary containment safety function was not in accordance with 10 CFR 50.72(b)(3)(v) and was a performance deficiency. Because violations of 10 CFR 50.72 are considered to be violations that potentially impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the Reactor Oversight Process (ROP) SDP. As such, a cross cutting aspect was not assigned to this violation. Per the NRC Enforcement Policy, Section 6.0, "Violation Examples," a failure to make a report required by 10 CFR 50.72 is categorized as a Severity Level IV violation.
Enforcement: Title 10 CFR 50.72 (b)(3)(v)(C) and (D) requires, in part, that licensees report any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to (C) control the release of radioactive material, and (D) mitigate the consequences of an accident.
Contrary to the above, March 3, 2011, the licensee failed to report a condition that could have prevented the fulfillment of a safety function. Specifically, in the case of secondary containment, when it was declared inoperable for failing to maintain greater than or equal to 0.25 inches of vacuum due to an open pathway, the secondary containment, by definition, was deemed incapable of performing its specified safety function. Because this violation was not repetitive or willful, and was entered into the licensee's CAP as AR 01182255 and AR 01195987, this violation is being treated as a Severity Level IV NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy

The associated performance deficienciy for the event is tracked as item 2011-002-02.
Inspection Report# : 2011002 *(pdf)*

Significance: SL-IV Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Design the ISFSI Pad to Adeequately Support the Static and Dynamic Loads of Stored Casks
The inspectors identified a NCV of 10 CFR 72.212 (b)(2)(i)(B), "Conditions of a General License Issued Under 72.210," for the licensee's failure to perform adequate evaluations of the ISFSI pad. Specifically, the inspectors identified five examples where the licensee failed to design the ISFSI pad to adequately support the static and dynamic loads of the stored casks, considering potential amplification of earthquakes through soil-structure interaction. The licensee documented the conditions in IR 900610, IR 966506, and IR 1102633. As an interim corrective action, the licensee provided a technical paper providing justification for partial loading of the pad with ten casks. The inspectors determined that the previously discussed examples were a violation that warranted a significance evaluation. Consistent with the guidance in Section 2.2 of the NRC Enforcement Policy, ISFSIs are not subject to the SDP and, thus, traditional enforcement will be used for these facilities. The inspectors determined that the violation was of more than minor significance because if left uncorrected, a failure of the ISFSI pad could lead to a

more significant safety concern. Consistent with the guidance in Section 2.6.D of the NRC Enforcement Manual, if a violation does not fit an example in the Enforcement Policy Violation Examples, it should be assigned a severity level:
(1) Commensurate with its safety significance; and (2) informed by similar violations addressed in the Violation Examples. The inspectors determined that the violation could be screened using Section 6.5.d.1 of the NRC Enforcement Policy as a Severity Level IV Violation.
Enforcement
Title 10 CFR 72.212 (b)(2)(i)(B) requires, in part, that the licensee perform written evaluations prior to use, that establish the cask storage pads and areas have been designed to adequately support the static and dynamic loads of the stored casks, considering potential amplification of earthquakes.
Contrary to the above, the licensee's completed evaluation did not adequately evaluate the cask storage pad to support static and dynamics loads of the stored casks considering potential amplification of earthquakes as demonstrated by the following examples:
1. Calculation L-003447, Revision 3 (8/17/2009), Dynamic Analysis of HI-STORM 100 Cask on LaSalle ISFSI pads: The inspectors identified that in lieu of performing a detailed dynamic analysis to determine seismic response of the cask, the licensee used the methodology described in the NUREG/CR6865. The inspectors determined that the calculation contained a number of assumptions and did not demonstrate the LaSalle ISFSI pad was bounded by the analyzed pad in NUREG/CR-6865.

2. Calculation L-003447, Revision 3 (8/17/2009), Dynamic Analysis of HI-STORM 100 Cask on LaSalle ISFSI Pads: The inspectors identified that the dynamic analysis did not capture three-dimensional effects, such as torsion, due to a partially loaded pad. The licensee failed to analyze the pad for the worst case cask configuration on the pad and thus failed to adequately address increased torsional dynamic responses on the pad.

3. Calculation L-003447, Revision 4 (5/12/2010), Final Design Basis Dynamic Analysis of LaSalle ISFSI Pad: The inspectors identified that the licensee used ASCE 4-98 as industry guidance for completion of the SSI. However, the licensee failed to address uncertainties in the soil in accordance with this standard. The omission reduced the licensee's calculated safety factor and should have been included in the licensee's analysis.

4. Calculation L-003447, Revision 4 (5/12/2010), Final Design Basis Dynamic Analysis of LaSalle ISFSI Pad: The inspectors identified that the licensee did not provide adequate justification and documentation for use of a new SSI analysis methodology.

5. Calculation L-003447, Revision 4 (5/12/2010), Final Design Basis Dynamic Analysis of LaSalle ISFSI Pad: The inspectors identified that the licensee's analysis used a single set of three-dimensional (two horizontal and one vertical) acceleration time-histories to complete the SSI analysis. The inspectors have determined that the licensee's use of only a single set of acceleration time-histories to perform a non-linear SSI analysis may have significantly underestimated the predicted seismic response and thus does not conservatively meet the requirements of 10 CFR 72.212.
This is a violation of 10 CFR 72.212 (b)(2)(i)(B), "Conditions of a General License Issued Under 72.210." This violation is being treated as a NCV consistent with Section 3.1.1 of the NRC Enforcement Manual. (NCV 05000373/2010005-03; 05000374/2010005-03; 07200070/2010-02, Failure to Design the ISFSI Pad to Adequately Support the Static and Dynamic Loads of Stored Casks). The licensee entered this violation into their corrective action program (IR 900610, IR 966506, and IR 1102633). This closes URI 07200070/2008001-01.

Inspection Report# : 2010005 *(pdf)*

Significance: SL-IV Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to perform adequate evaluations to ensure compliance with 10 CFR 72.212(b)(3) and 10 CFR 72.122 (b)(2)(i)
The inspectors identified a Severity Level IV NCV of 10 CFR 72.146, "Design Control," for the licensee's failure to perform adequate evaluations to ensure compliance with 10 CFR 72.122(b)(2)(i) and 10 CFR 72.212(b)(3). Specifically, the inspectors identified that the licensee failed to evaluate that the reactor site parameters, including analyses of tornado missiles, were enveloped by the transfer cask design basis and that the transfer cask was designed to withstand the effects of natural phenomenon including tornadoes. The licensee documented the conditions in IR

1137279 and initiated actions to evaluate the described condition.

Description

Title 10 CFR 72.122(b)(2)(i), “Overall Requirements,” states, in part, that “structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform their intended design functions.”

Title 10 CFR 72.212(b)(3), “Conditions of General License Issued Under 72.210,” states that the licensee shall “review the Safety Analysis Report (SAR) referenced in the Certificate of Compliance and the related NRC Safety-Evaluation Report, prior to use of the general license, to determine whether or not the reactor site parameters, including analyses of earthquake intensity and tornado missiles, are enveloped by the cask design bases considered in these reports. The results of this review must be documented in the evaluation made in Paragraph (b)(2) of this section.”

The Holtec FSAR Section 3.4.8.2, “HI-TRAC Transfer Cask,” Subsection 3.4.8.2.1, “Intermediate Missile Strike” states, in part, that the “HI-TRAC is always held by the handling system while in a vertical orientation completely outside of the fuel handling building. Therefore, considerations of instability due to a tornado missile strike are not applicable.” The Holtec FSAR did not evaluate the effects of a HI-TRAC tornado missile strike for overturning or sliding as it was determined by the CoC Holder to not be a credible event.

However, at the LaSalle County Station spent fuel storage processing operations are completed on the highest elevation floor of the Reactor Building, the refuel floor. While on the refuel floor, the HI-TRAC is not engaged to a handling system during processing operations. The Reactor Building siding and roofing on the refuel floor are designed to blow-in/blow-out or blow off at a predetermined wind pressure during a tornado event to protect the structural integrity of the structural steel, leaving an open pathway to the environment. Therefore, at the LaSalle County Station, during a tornado event on the refuel floor, there is a potential that tornado generated missiles and winds could impact structures, systems, or components, specifically the HI-TRAC.

During review of Calculation L-003400, “Decontamination Pit Grillage for Cask Loading – Reactor Building EL843,” Revision 1, and review of Calculation L-003498, “Tornado Evaluations for Byron, Braidwood, and LaSalle Station Dry Storage Projects,” Revision 0, the inspectors noted that the HI-STORM had been evaluated for the effects of a tornado while stored on the pad; however, the effects of a tornado were not addressed for the HI-TRAC while being processed on the refuel floor. The inspectors were concerned that the HI-TRAC was not analyzed for cask overturning or sliding due to a tornado generated missile strike or tornado wind pressure on the refuel floor.

The inspectors determined that the licensee failed to determine that the reactor site parameters, including analyses of effects of natural phenomenon including tornadoes, were enveloped by the cask design bases and subsequently failed to perform an additional analysis to ensure that the requirements of 10 CFR 72.122(b)(3) were met. Subsequent to the inspectors inquiry the licensee performed Calculation L-003582, “Tornado Analysis for LaSalle HI-TRAC,” Revision 0. Calculation L-003582 determined that overturning or sliding of the HI-TRAC at the refuel floor elevation would not occur due to the effects of a tornado. The inspectors reviewed the subsequent calculation.

Analysis

The inspectors determined that the licensee’s failure to perform a calculation evaluating the effects of a tornado on the HI-TRAC was a violation that warranted a significance evaluation. Consistent with the guidance in Section 2.2 of the NRC Enforcement Manual, ISFSIs are not subject to the SDP and, thus, traditional enforcement will be used for these facilities. The violation was determined to be of more than minor significance using IMC 0612, “Power Reactor Inspection Reports,” Appendix E, “Examples of Minor Issues,” Example 3i, in that the licensee’s lack of evaluation did not assure cask integrity during a design basis tornado and an additional calculation was required to evaluate the effects of the design basis tornado during canister processing operations in the Reactor Building refuel floor elevation in accordance with the ISFSI licensing/design basis analysis requirements.

Consistent with the guidance in Section 2.6.D of the NRC Enforcement Manual, if a violation does not fit an example in the Enforcement Policy Violation Examples, it should be assigned a severity level: (1) Commensurate with its safety significance; and (2) informed by similar violations addressed in the Violation Examples. The violation screened as having very low safety significance (Severity Level IV). Specifically, Calculation L-003582 determined that overturning and sliding of the HI-TRAC at the refuel floor elevation would not occur during tornado missile impacts.

Enforcement

Title 10 CFR 72.146(a), “Design Control,” states, in part, that “The licensee shall establish measures to ensure that applicable regulatory requirements and the design basis, as specified in the license for those structures, systems, and components to which this section applies, are correctly translated into specifications, drawings, procedures, and instructions. These measures must include provisions to ensure that appropriate quality standards are specified and included in design documents and that deviations from standards are controlled.”

Contrary to the above, on August 9, 2010, the licensee failed to establish measures to ensure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions. Specifically, the licensee failed to evaluate the effects of natural phenomenon, including tornadoes, on the HI-TRAC. This finding is being treated as a NCV, consistent with Section 3.1.1 of the NRC Enforcement Manual. (NCV 05000373/2010005-04; 05000374/2010005-04; 07200070/2010-03, Failure to Perform Adequate Evaluations to Ensure Compliance with 10 CFR 72.212(b)(3) and 10 CFR 72.122(b)(2)(i)). The licensee documented the violation in IR 1137279 and initiated actions to evaluate the described condition.

Inspection Report# : 2010005 *(pdf)*