STATE OF CALIFORNIA GAVIN NEWSOM, Governor

CALIFORNIA STATE LANDS COMMISSION
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Established in 1938

NOTICE OF DETERMINATION
(Public Resources Code section 21108)

File Ref: PRC 6785
SCH#: 2016071025
Item: 01

CEQA Project Title: San Onofre Nuclear Generating Station (SONGS) Units 2 & 3 Decommissioning Project

Lead Agency Discretionary Approval: Termination of existing California State Lands Commission (Commission or CSLC) General Lease – Industrial Use and Issuance of 16-year General Lease – Industrial Use through March 20, 2035 for use of CSLC Lease Facilities, including offshore riprap and CSLC Lease Offshore Activities associated with the Project.

Proponent: Southern California Edison Company, San Diego Gas & Electric Company, City of Riverside and the City of Anaheim

Location: The Project's offshore portion (CSLC Lease Offshore Activities) lies on State sovereign land under the jurisdiction of the Commission; the onshore Project site lies entirely within the boundaries of the Marine Corps Base Camp Pendleton, approximately 51 miles north-northwest of the city of San Diego, San Diego County, California.

Description: The Project involves decontamination, dismantlement, and removal of certain above- and below-grade facilities at SONGS to meet the Nuclear Regulatory Commission's requirements for unrestricted use; disposition of the Units 2 & 3 offshore conduits; and removal of navigational and environmental monitoring buoys and anchors. These activities would take place from 2019 through 2028. The portion of the Project within Commission jurisdiction is located on sovereign land and is defined as the CSLC Lease Offshore Activities; these activities include disposition of two offshore intake conduits and two offshore discharge conduits, and removal of two primary offshore intake structures, two auxiliary offshore intake structures, 12 diffusers, 23 manhole access port structures, one fish return conduit, and five monitoring and navigational buoys.

This is to advise that the California State Lands Commission, as ☒Lead Agency ☐ Responsible Agency, approved the above described Discretionary Approval associated with the project on March 21, 2019, and made the following determinations regarding the project.

1) The project ☒ will ☐ will not have a significant effect on the environment.
2) An ☒ Environmental Impact Report (EIR) was prepared and certified pursuant to the provisions of CEQA.
3) Mitigation measures ☒ were ☐ were not made a condition of the approval of the project.
4) A mitigation reporting or monitoring plan ☒ was or ☐ was not adopted for this project.
5) A Statement of Overriding Considerations ☒ was ☐ was not adopted for the project.
6) Findings ☒ were ☐ were not made pursuant to the provisions of CEQA.

The Environmental Impact Report (EIR) and Mitigation Monitoring Program prepared for this project by the Commission and certified on March 21, 2019, and the record of project approval, may be examined at the above shown Commission Sacramento office address.

DATE RECEIVED FOR FILING AND POSTING BY THE GOVERNOR'S OFFICE OF PLANNING AND RESEARCH

Governor's Office of Planning & Research

MAR 22 2019

STATE CLEARINGHOUSE

JENNIFER LUCCHESI, Executive Officer

Contact Person: Cheryl Hudson (916) 574-1900

State of California - Department of Fish and Wildlife
2019 ENVIRONMENTAL FILING FEE CASH RECEIPT
DFW 753.5a (REV. 12/01/18) Previously DFG 753.5a

Ck #073-1000168

Print | Start Over | Finalize&Email

RECEIPT NUMBER:
59 — 03/22/201 — 036

STATE CLEARINGHOUSE NUMBER *(If applicable)*
2016071025

SEE INSTRUCTIONS ON REVERSE. TYPE OR PRINT CLEARLY.

LEAD AGENCY	LEAD AGENCY EMAIL	DATE
CA State Lands Commission		03/22/2019

COUNTY/STATE AGENCY OF FILING	DOCUMENT NUMBER
OPR/SCH	

PROJECT TITLE

San Onofre Nuclear Generating Station (SONGS) Units 2 & 3 Post-Shutdown Decommissioning Project

PROJECT APPLICANT NAME	PROJECT APPLICANT EMAIL	PHONE NUMBER
Jennifer Lucchesi		(916) 574-1810

PROJECT APPLICANT ADDRESS	CITY	STATE	ZIP CODE
100 Howe Avenue, Suite 100 - South	Sacramento	CA	95825

PROJECT APPLICANT *(Check appropriate box)*

☐ Local Public Agency ☐ School District ☐ Other Special District ☑ State Agency ☐ Private Entity

CHECK APPLICABLE FEES:

☑ Environmental Impact Report (EIR)	$3,271.00	$	3,271.00
☐ Mitigated/Negative Declaration (MND)(ND)	$2,354.75	$	0.00
☐ Certified Regulatory Program (CRP) document - payment due directly to CDFW	$1,112.00	$	0.00

☐ Exempt from fee
- ☐ Notice of Exemption (attach)
- ☐ CDFW No Effect Determination (attach)

☐ Fee previously paid (attach previously issued cash receipt copy)

☐ Water Right Application or Petition Fee (State Water Resources Control Board only)	$850.00	$	0.00
☐ County documentary handling fee		$	
☐ Other		$	

PAYMENT METHOD:

☐ Cash ☐ Credit ☑ Check ☐ Other **TOTAL RECEIVED** $ 3,271.00

SIGNATURE	AGENCY OF FILING PRINTED NAME AND TITLE
X	Johnny Huynh

ORIGINAL - PROJECT APPLICANT | COPY - CDFW/ASB | COPY - LEAD AGENCY | COPY - COUNTY CLERK | DFW 753.5a (Rev. 12012018)

WM2019 Conference, March 3 – 7, 2019, Phoenix, Arizona, USA

Update on Development of a U.S. Rail Transport Capability for Spent Nuclear Fuel and High-Level Waste – 19092

Patrick Schwab*, Matt Feldman**, Bill Reich**, Steve Maheras***, and Scott Dam****
* U.S. Department of Energy
** Oak Ridge National Laboratory
*** Pacific Northwest National Laboratory
**** TechSource, Inc.

ABSTRACT

This paper provides an overview of the progress to date, and discussion of the path forward, related to designing, fabricating and testing prototype railcars that will comply with the safety standard S-2043. This standard was developed by the Association of American Railroads (AAR) specifically for railcars used to transport High-Level Radioactive Material (HLRM). AAR defines the term HLRM to include both spent nuclear fuel (SNF) and high-level radioactive waste (HLW). DOE is in the process of developing and testing prototype railcars that will satisfy Standard S-2043.

This is a technical paper that does not take into account the contractual limitations under the Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste (Standard Contract) (10 CFR Part 961). Under the provisions of the Standard Contract, DOE does not consider spent nuclear fuel in multi-assembly canisters to be an acceptable waste form, absent a mutually agreed to contract amendment. To the extent discussions or recommendations in this paper conflict with the provisions of the Standard Contract, the Standard Contract provisions prevail.

INTRODUCTION

The Association of American Railroads (AAR) has published a technical standard developed specifically for railcars used to transport High-Level Radioactive Material (HLRM): *Performance Specification for Trains Used to Carry High-Level Radioactive Material, Standard S-2043*. [1] AAR defines the term HLRM to include SNF and HLW. DOE is in the process of developing prototype railcars that will be compliant with S-2043.

Moving HLRM from its current locations will require both a receiving site or multiple sites and a transportation capability. The U.S. currently has the capability to move large rail-size casks of HLRM; however, improved rail transportation capability will be needed to move the HLRM in compliance with S-2043.

The total project to design and develop railcar prototypes, conduct the necessary testing, and secure approval of S-2043–compliant railcars is scheduled to be finished in 2022. Approval by the AAR will require extensive full-scale testing of the individual railcars and the complete rail consist.

HLRM will need to be shipped in transport casks certified in accordance with 10 CFR Part 71 by the Nuclear Regulatory Commission (NRC). The NRC has certified numerous rail transport cask designs supplied by various manufacturers. These rail transportation casks will weigh between approximately 82 and 210 tons when loaded; additionally, each cask, if transported by rail, would need to be attached to the railcar by a cradle (often called a “skid”) that is expected to weigh between 10 and 20 tons. No existing railcars have been approved as AAR S-2043 compliant for shipping these NRC-certified casks.

Therefore, transport of HLRM over the railroad infrastructure in the U.S. in railcars that meet S-2043 will require new railcars to be designed, tested, and approved by the AAR for use.

DOE-NE is currently funding efforts to have industry design, develop, fabricate, and test S-2043 compliant cask, buffer, and escort railcars. The result of this effort will be fabricated prototype railcars that will be tested to obtain approval from the AAR for transport of HLRM.

DOE is pursuing the design and development of S-2043 compliant railcars to help ensure that a capable transportation system will be available for the safe, secure, and efficient movement of HLRM from commercial nuclear power reactor sites and/or DOE sites in a timely manner when a receiving site becomes operational.

ATLAS RAILCAR DESIGN PROJECT OVERVIEW

The DOE awarded a contract on August 21, 2015 for the design, associated analysis, and prototype fabrication of cask and buffer railcars to transport HLRM. This contract was awarded to Orano Federal Services (OFS), with primary subcontractors Kasgro Rail and the Transportation Technology Center, Inc. (TTCI).

DOE awarded a second contract on June 27, 2018, for the single-car and multiple-car testing of the Atlas and buffer railcars. The railcar testing contract was awarded to the TTCI, with primary subcontractors Kasgro Rail and OFS. So the primary membership of the contractor team is unchanged from the first contract to the second one.

The new cask railcar design is named *Atlas*. The project's five phases are being implemented via two multi-year contracts. The Orano contract is for Phase 1—Mobilization and Conceptual Design, Phase 2—Preliminary Design, and Phase 3—Prototype Fabrication and Delivery. The TTCI contract is for Phase 4—Single-Car Testing and Phase 5—Multiple-Car Testing. A description of the major activities in each of the five project phases is provided in this paper. The summary-level project schedule is shown in Fig. 1.

Phase 1 conceptual designs were completed in October 2016 [2]. This work included preparation of planning documents for project management, requirements, and engineering, as well as development of cask loading procedures, conceptual design of cask cradles, and preparation of conceptual designs for the cask and buffer railcars.

Phase 2 preliminary designs were completed in March 2018 [3]. This phase included performing the required design analysis, modeling, and simulation of the railcar designs; preparing preliminary designs of the Atlas and buffer railcars; submitting a preliminary S-2043 design package to the AAR for review; and obtaining approval of that design package from the AAR. Updates to some Phase 1 documentation were necessary due to the addition of the HI-STAR 190SL and HI-STAR 190XL casks.

Phase 3 was started in October 2017 and will result in the prototype fabrication and delivery of one Atlas railcar and two buffer railcars by March 2019. These prototype railcars will be used in the Phases 4 and 5 testing activities.



Fig. 1. Schedule for Atlas Railcar Design Project

Phase 4 activities started in June 2018 and will include the acquisition of test and ballast loads; fabrication of instrumented wheel sets (IWSs) to support railcar dynamic testing; and various static, structural, and dynamic tests of the individual railcars.

Phase 5 will begin in the fall of 2020, to include testing of the complete rail consist, testing of the safety monitoring system, and a demonstration run. The full consist will include a cask railcar, two buffer railcars, and a rail escort vehicle (REV). The REV is where the armed security personnel will ride and watch over the whole train.

Fortunately, the U.S. Naval Nuclear Propulsion Program (NNPP) began designing a new REV a few years ago and the two federal agencies have now agreed to use the same REV design. This agreement was formalized in a Memorandum of Understanding [4] in May 2017, and the NNPP is responsible for single-car testing of this REV design. DOE will procure one REV unit for the purpose of testing the full consist during Phase 5.

PHASE 1: RAILCAR MOBILIZATION AND CONCEPTUAL DESIGN

The mobilization part of Phase 1 included typical project management planning activities such as development of a project management plan and other associated plans. Conceptual designs were developed for the cask cradles, cask railcars, and buffer railcars. General loading procedures were developed for each cask and cradle combination. Phase 1 was completed in October 2016 and the Phase 1 Conceptual Design Report is Reference 2.

The Atlas railcar is being designed to carry 17 different HLRM casks. During transport, a transportation cask must rest on a cradle, often called a skid, on top of the cask railcar deck.

The cask railcar is being designed to transport one cask at a time along with one cask cradle. The cask railcar design includes all needed attachment points and a description of how to attach each of the cradles to the deck.

OFS is responsible for interfacing with all the transportation cask vendors to obtain the transportation cask information necessary to design the cask railcar. One result of this collaboration has been the development of conceptual designs for cradles that accommodate each of the 17 casks. These conceptual cradle designs are used to determine the height of the cask's center of gravity above the railcar deck, the weight on each axle, etc., as necessary to perform the analysis and provide simulated cradle test weights and supporting information needed for testing of the railcar. These conceptual cradle designs, however, will not be carried through to preliminary designs or final designs. DOE will use simulated cradles during the testing phases of this project.

Cask Railcar Conceptual Design

The Atlas Railcar design shown in Fig. 2 is similar to the Navy's M-290 cask railcar design, the only other railcar to have achieved approval from the AAR to transport HLRM in compliance with S-2043. Both of these railcars have twelve axles. The major components under the railcars' decks, including the trucks and tri-span bolsters, are virtually identical. The decision to use the design experience from the Navy's program greatly increases the probability of success in the Atlas Railcar Design Project.

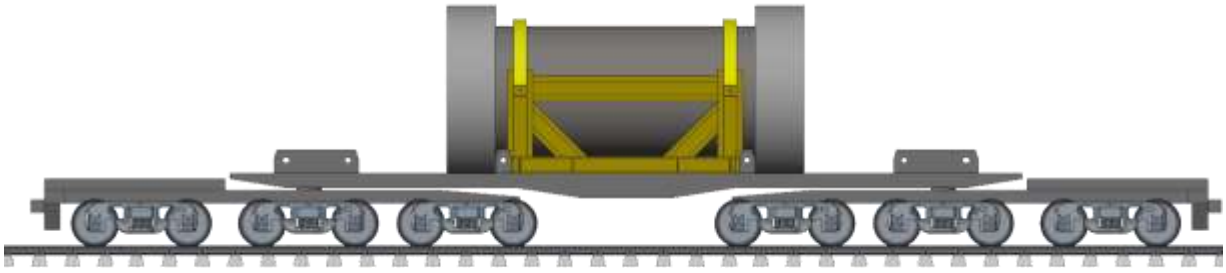


Fig. 2. Atlas Railcar Conceptual Design

Buffer Railcar Conceptual Design

Buffer railcars are flatbed cars that are required to separate HLRM cask cars from the cars carrying people (locomotives and REVs). The buffer railcar conceptual design is a four-axle freight railcar as shown in Fig. 3. The process of developing a conceptual design calls for generating basic design information such as dimensions, deck layout, and number and type of trucks and then providing that information to the simulation team for the initial design analysis and modeling that occurs in Phase 2.



Fig. 3. Buffer Railcar Conceptual Design

This buffer railcar design uses the same trucks as the REV, which is also designed to satisfy AAR Standard S-2043. This reduced the number of issues that could have occurred during the design analysis and preliminary design development.

Atlas Railcar Heaviest Load

The Atlas railcar carrying the Holtec HI-STAR 190XL cask is shown in Fig. 4. This is the largest cask payload that the Atlas railcar is designed to carry.

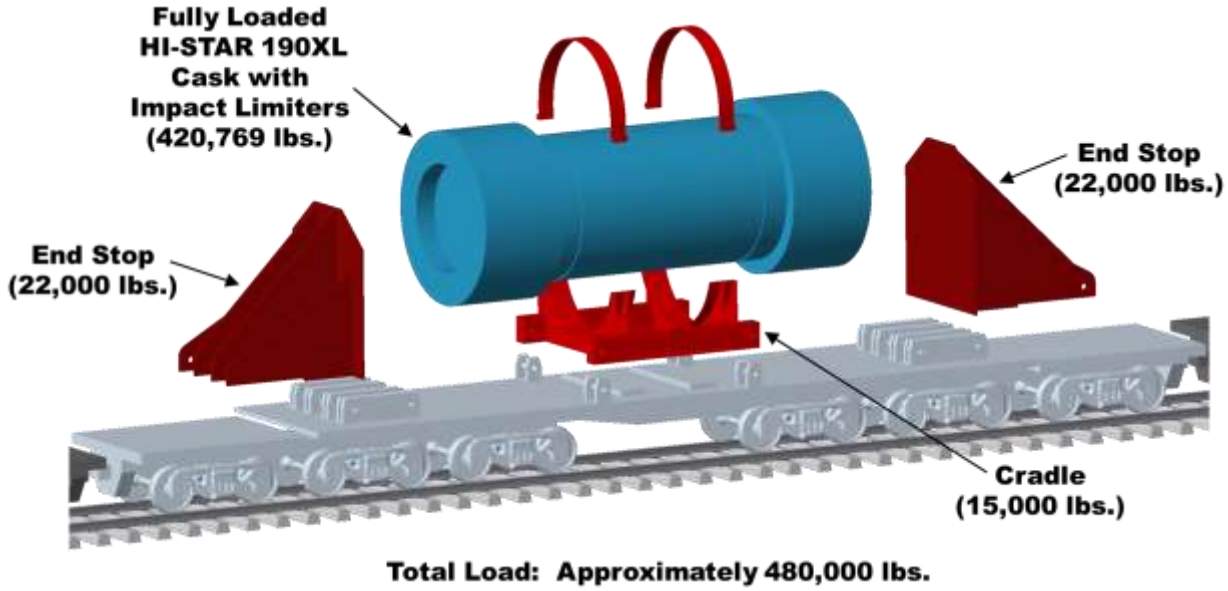


Fig. 4. Atlas Railcar with HI-STAR 190XL cask

PHASE 2: RAILCAR PRELIMINARY DESIGN

The Phase 2 preliminary design activities were completed in March 2018. This includes performing the required design analysis, modeling and simulation of the railcar designs, preparing preliminary designs of the Atlas and buffer railcars, and submitting a preliminary S-2043 design package to the AAR for review and approval.

The conceptual designs from Phase 1 were developed further in Phase 2 to include detailed technical descriptions of the railcars, general design drawings, parts lists, and detailed documentation of the design analysis, modeling, and simulation. Phase 2 resulted in a detailed preliminary design package with sufficient information to support fabrication of prototype Atlas and buffer railcars in Phase 3.

Design Development, Analysis, Modeling, and Simulation

The railcars must be designed to meet AAR Standard S-2043 design requirements in six areas: (1) structural analysis, (2) nonstructural static analysis, (3) dynamic analysis, (4) brake system design, (5) system safety monitoring, and (6) railcar clearance and weight.

Structural analysis of the railcar designs is performed using commercial finite-element analysis (FEA) software that considers a number of load and force combinations, including dead load, live load, coupler load, compressive end load, and impact load as specified in S-2043. The analysis includes an assessment of whether the cask cradles and securement points can withstand specific static, cyclic, and dynamic loads. There are also crashworthiness requirements to ensure that railcar coupling systems, trucks, and wheel sets will not separate from one another in the event of a derailment.

Nonstructural static analysis includes an assessment of truck twist equalization to ensure adequate truck load distribution under statically applied track twist conditions, car body twist equalization to document the amount of wheel unloading during car body twist, like that encountered during negotiation of a spiral, and static curve stability analysis to calculate the amount of wheel unloading in adverse curving scenarios.

Dynamic analysis simulations of the railcars are modeled for 14 areas of track performance. The purpose of the simulations is to provide a realistic basis for evaluation of railcar dynamic performance under less-than-ideal conditions. This demonstrates that the railcar designs provide an adequate margin of safety from structural damage and from any tendency to derail. Each simulation must meet nine performance criteria related to lateral and vertical wheel forces, car body roll angle, and various car body accelerations.

The brake monitoring system transmits vital safety-related information (e.g., brake pressure, roller bearing temperature, vibration level, etc.) to the locomotive, which will allow the train crew to stop the consist in the event of equipment malfunction or failure. The brake design is modeled during the dynamic simulations.

The railcar design includes a reliable and robust safety monitoring system designed to prevent derailments caused by equipment degradation or failure. The system is designed to provide the train crew with real-time monitoring of 11 performance parameters, including railcar location, speed, truck hunting, rocking, wheel flats, bearing condition, ride quality, braking performance, vertical acceleration, lateral acceleration, and longitudinal acceleration.

The railcar designers provided railcar clearance diagrams for each railcar design and determine bridge loadings for each railcar.

Preliminary S-2043 Design Package Submittal to AAR

The railcar designer submitted a preliminary S-2043 design package to the AAR Equipment Engineering Committee (EEC) in July 2017. The EEC performed an in-depth technical review of the railcar designs which is required prior to Phase 3 prototype fabrication.

DOE received the EEC's notification to proceed with the railcar test phase on February 2, 2018, which allowed the railcar designer to proceed with the Phase 3 fabrication of the railcar prototypes.

PHASE 3: PROTOTYPE RAILCAR FABRICATION

Phase 3 includes the prototype fabrication of one Atlas railcar and two buffer railcars. This phase is scheduled to be completed in March 2019. These prototype railcars will be used in the Phases 4 and 5 testing activities.

Material Procurement

Raw material procurement continued throughout FY 2018 in support of the fabrication of the prototype Atlas and buffer railcars. Kasgro Rail continues to perform quality receipt and acceptance inspections on all materials received. OFS provides QA oversight of the inspection process.

Fabrication Mobilization

Kasgro Rail began and completed the necessary setup at its facilities earlier this fiscal year in preparation for fabrication of the railcars. A special area was established at Kasgro Rail to store smaller Atlas railcar components (e.g., bolts, nuts) to ensure that no components used for Atlas railcar fabrication are intermingled with components from other fabrication activities, since the components for Atlas railcars have been procured only from vendors on the Kasgro Approved Vendors List.

Component Procurements

While many railcar components will be fabricated at the Kasgro Rail facility, some are being procured from rail industry vendors. Most notably, the wheel sets, truck assemblies, and brake systems are being procured from other vendors. The safety monitoring system for the railcar consist, manufactured by Lat-Lon (www.lat-lon.com), was procured and received. All procurements were completed during FY 2018.

Wheel Sets and Truck Assemblies

A *wheel set* is an assembly that consists of two wheels fitted to an axle. This is an important component of the freight truck, since the interaction of the rail and wheel profiles is important to the stability and curving performance of the railcar. Two wheel sets connected by two sideframes and a center bolster are called a *three-piece freight truck* or *bogie*.

Rail industry partners that specialize in providing wheel sets and complete truck assemblies are providing these components for the Atlas and buffer railcars.

Railcar Fabrication

Kasgro Rail is fabricating one prototype Atlas railcar and two prototype buffer railcars in accordance with the AAR-approved preliminary designs developed in Phase 2 of this project. One of the completed buffer railcars is shown in Fig. 5, with the weights welded on top of the buffer railcar clearly visible.



Fig. 5. Buffer Railcar

Kasgro has nearly completed the fabrication of the Atlas railcar. Fig. 6 shows this railcar from two different perspectives as of September 2018. The final inspection of all three prototype railcars is scheduled for February 2019.

Railcar Final Delivery

One prototype Atlas railcar and two prototype buffer railcars are scheduled to be delivered to AAR's Transportation Technology Center near Pueblo, Colorado, in March 2019. TTCI will then begin single- and multiple-car testing of the railcars. OFS will deliver the as-built Atlas Railcar Design Project documentation package to DOE as part of the final Phase 3 Report.



Fig. 6. Atlas Railcar as of September 2018

ACQUISITION OF RAIL ESCORT VEHICLE

The REV is a passenger railcar that will provide conveyance for armed escorts as required in 10 CFR 73.37 for rail shipments of irradiated reactor fuel. The REV must be included in the consist during the multiple-car testing to meet all S-2043 requirements. DOE considered the options available to acquire an REV and decided to use the design already being developed for the NNPP. The NNPP REV design will likely meet all AAR and DOE requirements to provide effective physical protection for rail shipments of HLRM.

DOE and the NNPP negotiated and signed a memorandum of understanding (MOU) on May 15, 2017 that describes DOE's intent to adopt the NNPP REV design. DOE and the NNPP signed an Agreement and Project Plan for Rail Escort Vehicle Procurements on June 13, 2018. The NNPP is responsible for single-car testing of the REV design, then DOE will use these results in its multiple-car testing program. The NNPP plans to use the DOE buffer car design.

The NNPP plans to procure a total of five REVs. The NNPP will acquire the first REV manufactured, and DOE will acquire the second REV. The contract to fabricate DOE's REV is scheduled to be signed in January 2019, and the actual fabrication should start in April 2019. DOE's REV is scheduled to be delivered to TTC in September 2020, just before the start of DOE's multiple-car testing. The NNPP will then procure the next four REVs. The first REV during fabrication is shown in Fig. 7.



Fig. 7. Fabrication of a Rail Escort Vehicle

PHASE 4: SINGLE-CAR TESTING

AAR Standard S-2043 requires completion of a comprehensive set of single-car tests to confirm the results of the design simulations and to ensure that the railcars meet the stringent safety requirements of the standard. The single-car testing is conducted in five major categories: (1) vehicle characterization, (2) nonstructural static tests, (3) static brake tests, (4) structural tests, and (5) dynamic tests.

Vehicle characterization is performed to verify that the components and vehicle as a whole are built as designed. This involves characterizing properties of the railcar body and its suspension using off-track testing methods. Characterization results will aid the railcar designers in estimating performance under conditions that cannot easily be tested. Tests include system-level and component-level stiffness, damping and resonances, twist and roll, pitch and bounce, yaw and sway, and hunting performance. Other tests in this category include truck rotation stiffness and break-away moment, inter-axle longitudinal stiffness, and axle yaw stiffness.

Nonstructural static tests are performed to demonstrate the ability of the railcar to maintain adequate vertical wheel loads in extreme load conditions and poor track geometry environments. Tests include truck twist equalization, car body twist equalization, static curve stability, and horizontal curve negotiation.

Static brake tests include measurement of braking ratios of the braking systems on the railcar consist. Braking ratio is the braking force of a full brake application of a railcar divided by the total weight of the railcar.

Structural tests are conducted to demonstrate the railcar's ability to withstand the rigorous railroad load environment and to verify the accuracy of the structural analysis modeling. Tests include compressive end loading, coupler vertical loads, jacking, twisting, and impact tests. A minimum of 50 strain gages on the railcar underframe must be monitored and compared to the Finite Element Analysis (FEA) predictions.

Dynamic tests are conducted and the results compared to the dynamic analysis simulations previously performed. Instrumented wheel sets are required and are used to measure dynamic performance in 14 areas of track performance under less-than-ideal conditions. This demonstrates that the railcar designs provide an adequate margin of safety from structural damage and from any tendency to derail. Instrumented wheel sets are railway wheel sets that have been instrumented and calibrated so that they are capable of accurately measuring the dynamic contact forces between the wheel and rail.

Design and Fabrication of Modular Test Loads

OFS has designed test and ballast loads that will support the Phase 4 and 5 testing. A modular test load design was developed to allow TTCI to use its existing cranes to lift and place the various test loads on the Atlas railcar for testing. The modular design will allow the test loads to add ballast to an empty railcar and to simulate the weights of various sizes of casks, various sizes of cask cradles, and cask end stops. The test loads simulate the components' (casks, cradles, and end stops) weight, center of gravity, combined center of gravity, and mass moment of inertia.

Three test load configurations will be used during the railcar testing:

1. Empty railcar configuration: Atlas railcar + ballast test load
2. Minimum load configuration: Atlas railcar + empty small cask + cradle
3. Maximum load configuration: Atlas railcar + loaded large cask + cradle + end stops

During preliminary design of the Atlas railcar, it was determined that a ballast load of 200,000 pounds is required for the Atlas railcar to meet AAR S-2043 guidelines when transported empty in an S-2043 consist.

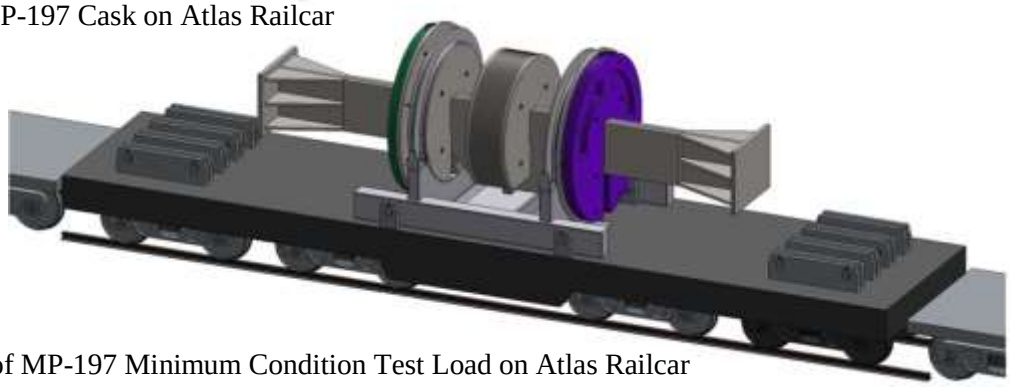
The minimum test load configuration simulates the Atlas railcar loaded with an empty small cask and cradle, as shown in Fig. 8. This figure depicts what an actual empty small cask would look like loaded on the Atlas railcar (upper left in figure) and also shows the modular test load that simulates this configuration.

The maximum test load configuration simulates the Atlas railcar loaded with a large cask full of HLRM, a cask cradle, and end stops as shown in Fig. 9. The figure depicts what an actual loaded large cask would look like when placed on the Atlas railcar (upper left in figure) and shows the modular test load that simulates this configuration.



Total weight on rails = 430,000 lbs.

Depiction of MP-197 Cask on Atlas Railcar



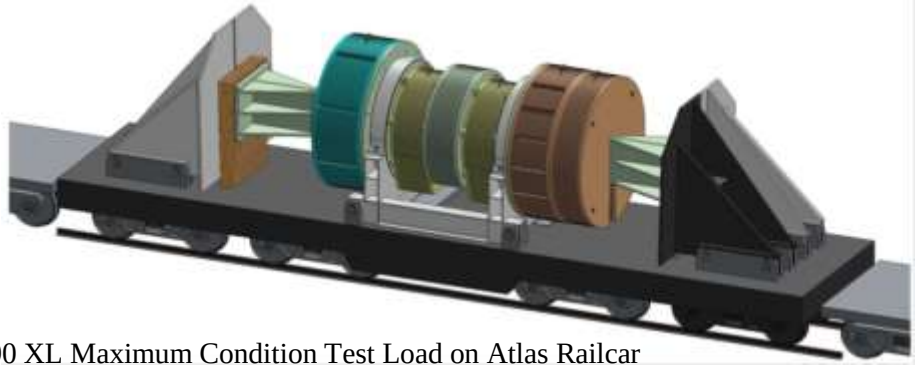
Depiction of MP-197 Minimum Condition Test Load on Atlas Railcar

Fig. 8. Minimum Railcar Test Load



Total weight on rails = 703,000 lbs.

Depiction of HI-Star 190 XL on Atlas Railcar



Depiction of HI-Star 190 XL Maximum Condition Test Load on Atlas Railcar

Fig. 9. Maximum Railcar Test Load

PHASE 5: MULTIPLE-CAR TESTING

The multiple-car test train consist must match the anticipated DOE HLRM train as closely as possible with a minimum of one of each type of car. It is anticipated that a DOE consist will include two locomotives, followed by one buffer car, then one or more Atlas railcars, another buffer railcar, and finally one REV as shown in Fig. 10.



Fig. 10. Complete HLRM Rail Consist

The multiple-car tests are designed to verify that the individual cars do not adversely affect the performance of adjacent cars. There are three categories of multiple-car tests, including dynamic, system monitoring, and revenue service tests.

Dynamic tests are performed to measure braking stop distance under various loads and rail conditions, braking performance in curves, ability of the hand brakes to hold the train on a grade, and buff and draft loads in curves.

System monitoring tests are conducted on all the train's communications and monitoring systems under normal service conditions and during failure simulations of various components.

Revenue service tests are performed to demonstrate acceptable performance under revenue service conditions such as turnouts, crossovers, and on poorly maintained spur tracks.

The final activity required by S-2043 is a demonstration run, as specified in paragraph 6.3.4 of the standard. The demonstration run must be performed over an actual rail service route. No special instrumentation or measurements are required during the demonstration. An operator's log must be kept throughout the trip to document train operation and any problems that occur. The log, combined with a route map and photo documentation, must be submitted as part of the final testing report described in the following paragraph.

The Atlas railcar final testing report will be completed in Phase 5 of this project and submitted to the EEC for a final review with a successful outcome resulting in approval to operate the railcars. After satisfactory operation of the railcars for approximately 100,000 service miles, the railcar designer will submit a follow-up test report to the EEC for final review. A successful outcome would result in the EEC granting full approval of the Atlas railcar design to DOE.

OPERATIONAL CONSIDERATIONS

After AAR approval of the Atlas railcar design, the railcars will be used during transportation operations for HLRM. Standard rail industry operating practices will be followed, as detailed in Federal Railroad Administration and AAR standards, guides, and field manuals. However, there are also special operating standards and requirements for the Atlas consist, as specified in AAR Standard S-2043, Appendix B: "Operating Standard for Trains Used to Carry High-Level Radioactive Material." This appendix establishes special operating standards for these railcars, including the requirement to follow the latest revision of AAR Circular Letter OT-55: "Recommended Railroad Operating Practices for Transportation of Hazardous Materials." The appendix states that all pre- and post-loading inspections must be completed and that all system safety monitoring equipment must be functioning correctly before train movement. The appendix also includes special crew training requirements specific to railcars used to ship HLRM.

DOE will have several options to choose from (e.g., lease, purchase) to establish and maintain the necessary railcar fleet for HLRM shipments. Once the prototype railcar testing is under way, DOE will decide whether to procure or lease the necessary rolling stock. DOE does not plan to develop any new locomotives but will instead expect private railroad companies to provide locomotives compatible with the S-2043–compliant railcars.

CONCLUSIONS

Significant progress has been achieved on the Atlas Railcar Design Project over the last year. Prototype fabrication of the cask railcar and two buffer railcars is nearing completion. DOE and the NNPP signed an *Agreement and Project Plan for Rail Escort Vehicle Procurements* in June 2018. Fabrication of the DOE REV will begin this spring.

A contract was signed in June 2018 for the single-car and multiple-car testing of the prototype railcars. Modular test and ballast loads have been designed to support the railcar testing activities.

Over the last five years, DOE has been pursuing the design and development of S-2043 compliant railcars to help ensure that a capable transportation system will be available for the safe, secure, efficient, and timely movement of HLRM from commercial nuclear power reactor sites and/or DOE sites when a receiving site becomes available.

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Congress of the United States
Washington, DC 20515

April 1, 2019

The Honorable Marcy Kaptur
Chairwoman
House Appropriations Subcommittee on
Energy and Water Development
2362 Rayburn House Office Building
Washington, DC 20515

The Honorable Mike Simpson
Ranking Member
House Appropriations Subcommittee on
Energy and Water Development
1016 Longworth House Office Building
Washington, DC 20515

Dear Chairwoman Kaptur and Ranking Member Simpson:

We write to request your assistance in addressing a long-standing issue that exists in each of our districts and states – finding a path forward for the timely removal of spent nuclear fuel (SNF) and material classified as Greater-Than Class C waste (GTCC waste) from sites where nuclear power generating activities have permanently ceased.

It is well known that the federal government has been found by the courts to be in partial breach of the contractual obligation created by the 1982 Nuclear Waste Policy Act to remove this material beginning in January 1998. Over the course of litigation in the intervening years, the U.S. Court of Claims has consistently ruled in favor of companies for expenditures they have incurred in the continuing storage of these materials. The Department of Energy (DOE) has previously estimated that the resulting impact on taxpayers could reach or exceed $30.8 billion – assuming it is in a position to begin meeting its obligation through the use of a pilot consolidated storage facility in 2021. Any delay in this schedule will only increase the Department's liabilities. These funds are paid out of a permanent appropriations account known as the Judgment Fund and they add to the federal deficit without benefit of budget or appropriations considerations.

DOE has been obligated to draw from the Judgment Fund because issues associated with the management of SNF and GTCC material at permanently shuttered nuclear plants have been at a stalemate for an extended period of time. We were pleased to see that these issues were a major focus of the Blue Ribbon Commission on America's Nuclear Future during the Obama Administration, and remain a top priority of the Trump Administration as well.

Moreover, the Trump Administration's fiscal year 2020 budget for the Department of Energy requests robust funding to address SNF and GTCC material via an interim storage program. The requested funding could help with the initiation of a consolidated interim storage (CIS) program within DOE with an initial focus on accepting SNF and GTCC waste from the shutdown reactors. The NRC has already docketed two CIS facility applications, which are now undergoing multiyear technical review processes to ensure the facilities would meet all applicable standards.

With this in mind, we respectfully request $25 million to support the development of a CIS program at DOE, complete the necessary applications, and assist with site preparation activities

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and regional transportation efforts of SNF. Specifically, we request $10 million for the initiation of a robust CIS program at DOE; $10 million for site preparation activities at Stranded Fuel sites preparing to move SNF to interim storage facilities; and $5 million to support DOE's efforts to reinitiate Regional Transport compacts and transportation coordination.

Our constituents have waited patiently for action on this matter, and it is past time that we end the continued stalemate that is wasteful of taxpayer resources and detrimental to the redevelopment of these sites in our communities. We urge you to include funding for SNF and GTCC waste storage and disposal in order to move forward and make these important advances in the nation's used fuel management program.

Sincerely,

DORIS O. MATSUI
Member of Congress

CHELLIE PINGREE
Member of Congress

MIKE LEVIN
Member of Congress

TONY CARDENAS
Member of Congress

JOE COURTNEY
Member of Congress

SALUD CARBAJAL
Member of Congress

SCOTT PETERS
Member of Congress

JERRY MCNERNEY
Member of Congress

HARLEY ROUDA
Member of Congress

RICHARD E. NEAL
Member of Congress

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HI-STORM UMAX Canister Storage System
Docket # 72-1040

Incidence and Consequences of Surface Scratches on the MPC Shell

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A Public Presentation to the NRC

By: Charles Bullard & Dr. Stefan Anton
Holtec International (*Holtecinternational.com*)

April 10, 2019, White Flint, MD

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Presentation Agenda

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Areas of presentation:

- Consideration of Scratches in the ASME Code, regulatory and licensing space
- Information in the UMAX FSAR related to MPC scratches and the basis for change

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Principal Meeting Objective

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- To present Holtec's regulation-informed and ASME Code-based position on MPC surface scratches for NRC's consideration.
- To provide a definitive case why the statement in the Operations chapter (Section 9.5) of the "UMAX" FSAR on the plausible absence of scratches during Canister loading is **NOT** safety-significant information and does **NOT** affect functional performance under the Certificate-of-Compliance granted by the NRC.

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Origin and Prevalence of Scratches

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- Scratches in plates and shells in the manufacture of any weldment are present everywhere. From the time a stack of plates arrives at the factory, the rubbing between the plate layers causes scratches to develop. Most plates come from the mill bearing scratches. Rolling, bending, forming, shearing, machining, etc., necessary to fabricate are all potential scratch-making operations. Surface scratches are commonplace and expected in MPCs.
- Quality manufacturing plants, such as ours, place a limit on admissible depth and population density of scratches, primarily to address workmanship considerations. Holtec's procedure HSP-320 has been used to screen scratches in thousands of hardware supplied by the company to nuclear plants.

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The Many Incarnations of Scratches

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- Two materials rubbing against each other produce scratches through abrasive wear.
- Materials that have close metallurgical affinity (such as austenitic stainless against austenitic stainless) are also susceptible to adhesive wear (galling) which can cause the scratch to be exacerbated.
- The depth and width of a scratch depends on the contact force and the state of the armor on the stainless steel surface (which varies from plate to plate). However, *the greater the contact force, the deeper the scratch.*
- Predicting the size of the scratch, therefore, is not an exact science.

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Minimizing the Extent of Scratching

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- Minimizing the extent of scratching on the Canister's surface must occur in conjunction with other considerations.
- For example, the ID of the Shield Ring opening in "UMAX" is 1/2 inch larger than the Canister OD to minimize the crew dose; the MPC Seismic Restraints are likewise closely positioned.
- The result: UMAX has a very high seismic tolerance capability, and the dose sustained by the crew is negligible.
- Scratches, which are purely cosmetic, are tolerated for the sake of the far more important benefits mentioned above.

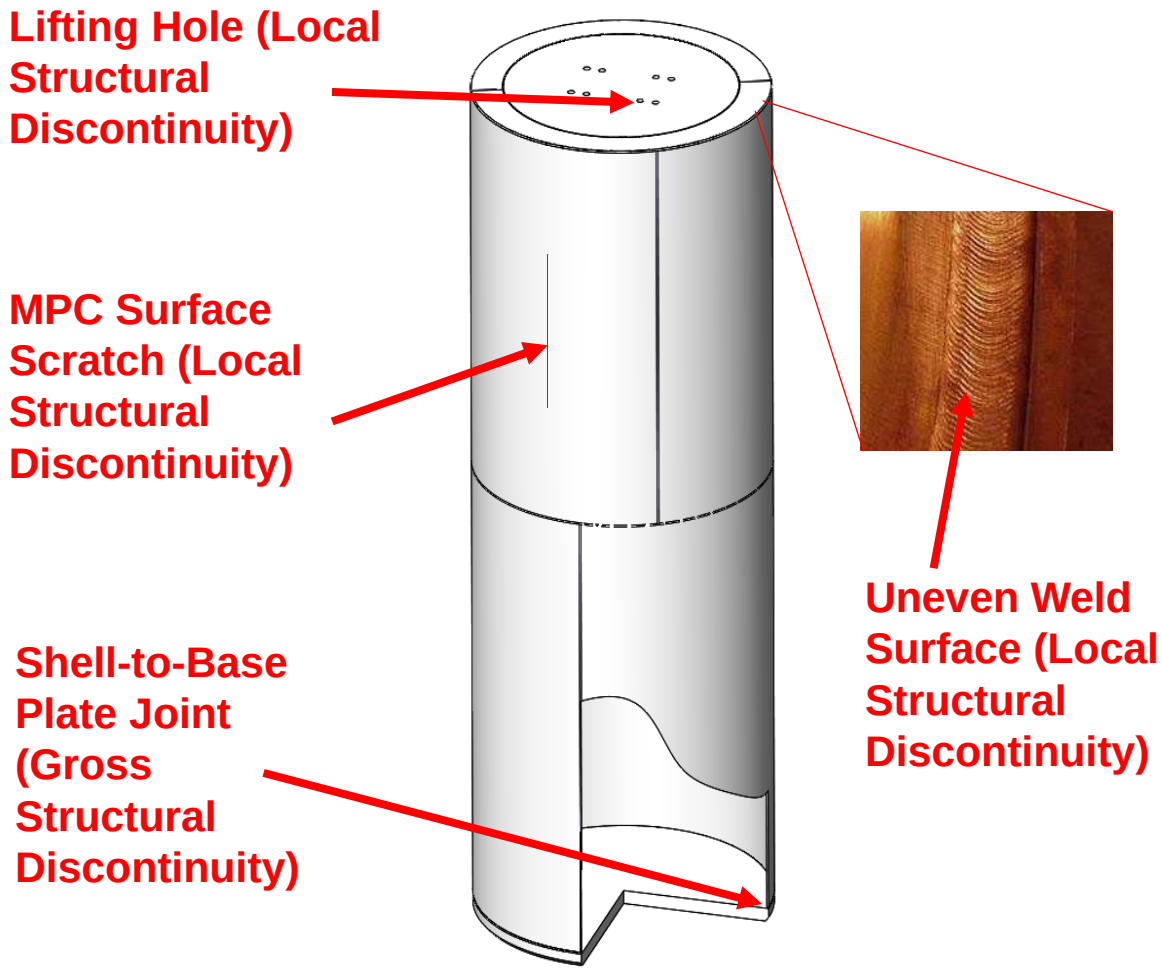
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The MPC is replete with local discontinuities such as scratches

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- As the figure shows, the Local Discontinuities are everywhere in a manufactured MPC.
- Every ridge and valley in an as-welded seam is a Local Discontinuity.



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Vertical Canister Insertion in "UMAX" minimizes scratches and their severity

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- Compared to horizontal insertion, where scratching occurs by the full weight of the Canister, the maximum contact load in UMAX, under the most limiting condition tolerance stack up, is merely ~ 2500 lbs. (2.5% of the Canister's weight).
- As a result, the scratches, if any, are inevitably shallow.
- Numerical analyses by Holtec showed the maximum depth of the scratch, due to abrasive wear, to be roughly 1.2 mils (0.0012") per 1000 lbs. of force.
- Tests in the shop to simulate the MPC shell contact showed a deeper scratch (but still quite shallow) because the scratching was accompanied by adhesive wear (i.e., galling of the plate surface).

Existing surface scratches do not impact the safety and functionality of the MPC.

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Quantifying the Maximum Scratch Depth

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- Shop tests and finite element analyses showed that the maximum average scratch depth (due to abrasive wear) is bounded by 2.4 mils (0.0024”).
- Minimum wall thickness required to meet the design internal pressure for the Canister is only 0.216” (per NB-3324 of the ASME Code); leaving a substantial thickness reserve (> 0.25”) for imperfections such as scratches.
- Extensive examination of the loaded Canisters has showed most loading related scratches to be invisibly small; the average depth across the width the scratch mark was near zero, and the deepest valley at any measured scratch location was 26 mils (0.026”) deep.

The scratches introduced during insertion remain a fraction of the required limit in the ASME Code.

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How does the ASME Code classify surface scratches in an MPC?

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- The MPCs are designed to meet the highest pedigree of the ASME Code, Section III Subsection NB, to the maximum extent practicable. The MPCs are manufactured to the same rigorous requirements as the Reactor Vessel and the Pressurizer in nuclear reactors.
- Several types of stresses are produced in a pressure vessel; they are classified by their degree of significance to the equipment's safety:
  - Primary Stress (NB-3213.8): Controlling, it is most significant to equipment's safe operation.
  - Secondary Stress (NB-3213.9): Somewhat significant, produced at locations of ***gross structural discontinuity.***
  - Peak Stress (NB-3213.11): Non-significant in quiescent equipment such as an MPC; develops at locations of ***local structural discontinuity***.

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Import of Local Discontinuities in a Canister

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- Examples of local discontinuities are: unevenness in the welded mass, scratches on the surface, vent and drain port holes, non-linear component of thru-thickness stress distribution from pressure & other loadings.
- Every MPC has local discontinuities whether there are scratches or no scratches.
- Because the local discontinuities produce only peak stresses which are inconsequential to safety, they are not considered in the FSARs.
  - Table 3.1.10 of the HI-STORM UMAX FSAR describes Peak Stress (F) as follows:

    "Increment added to primary or secondary stress by a concentration (notch), or, certain thermal stresses that may cause fatigue but not distortion. Because fatigue is not a credible source of failure in a passive system with gradual temperature changes, the cumulative damage factor from fatigue is not computed for HI-STORM UMAX components."

Local Discontinuity produces peak stress which is immaterial to the safety margin in an MPC

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Code places no limit on peak stresses

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- Local discontinuities and hence peak stresses are ubiquitous in every MPC. The Code places no limit on their number or size because they are irrelevant to the vessel's safe operation.
- That is why Chapter 2- Principal Design Criteria in the UMAX FSAR has no limits on peak stresses.
- That is why the NRC's Safety Evaluation Report (SER) makes no mention of the Commission having considered the peak stresses.
- That is why the Certificate-of-Compliance makes no mention of any type of local discontinuity (such as scratch) or peak stresses in the MPCs.

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Classifying Contents of a FSAR

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> The FSAR's content can be divided in three categories:

1. Safety critical: This material is placed and locked down in the CoC. It can only be altered by a License amendment.

2. Safety significant (SS): Not locked down in the CoC but subject to a highly controlled change process under 10CFR72.48.

3. Non-safety significant (NSS): Supplemental information that is not relied on to fulfill a safety function and is not included in the NRC SER.

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Regulatory Perspective on Scratches on Canisters

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- The scratches are not mentioned or relied on in the NRC's SER for "UMAX".
- Indeed, our research shows, scratches have never been considered a safety issue in any SER written by the NRC for any Holtec storage system.
- The NRC treated scratches in Canisters as non-germane to safety going back to the 1980s.

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Regulatory and Licensing Perspectives

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- From a regulatory perspective, the presence of surface scratches on the MPC are not relevant to the safety determination.
- Any discussion of scratches in the FSAR represents non-safety significant information, along with numerous other non-safety items in the MPC.
- Given that information in the FSAR on surface scratches is non-safety significant, clarifying the associated FSAR discussion cannot, by definition, imply a change of any safety consequence.
- The statement in the FSAR regarding no risk of scratching during insertion can be deleted or amended without affecting the safety analyses and conclusions in the FSAR.

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Licensing Perspectives

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> Filing a License Amendment Request (LAR) that does not pertain to a safety consideration would be a *non sequitur*.

> We have accordingly changed the following statement in Section 9.5:

"Because the MPC insertion (and withdrawal) occurs in the vertical configuration with ample lateral clearances, there is no risk of scratching or gouging of the MPC's external surface (Confinement Boundary). Thus the ASME Section III Class 1 prohibition against damage to the pressure retaining boundary is maintained."

to (through the 72.48 process):

"MPC insertion and withdrawal occurs in a tightly controlled vertical configuration with minimal lateral clearances. The MPC may come in contact with the shield ring and upper and lower seismic restraints (when present). The confined vertical configuration during MPC insertion limits the lateral loads placed on the MPC and precludes significant wear or scratches on the MPC's external surface (Confinement Boundary). Because scratches produce only localized stress raisers known as "peak stresses" in Section III of the ASME Code which (as noted in Table 3.1.10 of this FSAR) are germane only to cyclic fatigue considerations, the applicable stress intensity limits of ASME Section III Class 1 (primary and secondary) for the pressure retaining boundary will remain unaffected by the presence of scratches."

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Licensing Perspectives (cont.)

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- FSAR Section 9.5 was only meant to summarize (using several high level narrative examples) why Safety and ALARA is achieved through the operational steps outlined in Sections 9.1 - 9.4, as well as through the HI-STORM UMAX design itself.
- Thus, Section 9.5 simply makes a conclusion based on the preceding sections. It does not, nor was it intended to, articulate new or additional safety or operational requirements. This fact is supported by the omission of Section 9.5 from all other Holtec ISFSI FSARs.

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Licensing Perspectives (cont.)

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- The CoC/TS references the FSAR in various locations, for example in Table 3-1 with respect to ASME Code.
- When statements in the FSAR are incorporated into the CoC/TS in the sense that they become a legal part of it, they are identified clearly to avoid any misunderstanding. Examples can be found in the HI-STORM 100 FSAR, Section 9.1.5.3, and HI-STAR 180 SAR, Section 1.2.2.1, UMAX CoC Appendix B, Section 5.3.9. A general reference to the FSAR in the CoC is therefore not considered an inclusion by reference. Hence independent of the discussion about the relevance of Section 9.5 above, the statement about “no risk of scratching” can not be considered as being included by reference in the CoC.

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Closing Remarks

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- Localized scratches are an inevitable factor in MPC manufacture, transport and operation.
- Localized scratches in the MPC are not of concern from the standpoint of compliance with ASME Section III, Subsection NB stress intensity limits, nor do they violate any design or licensing requirements per the HI-STORM UMAX FSAR.
- Localized scratches in the MPC are not a safety concern nor do they affect its functionality.
- The discussion of scratches in FSAR Section 9.5 represents non-safety significant information.
- Using 10CFR72.48, Holtec has clarified the FSAR language describing wear and scratches on the MPC external surface.

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End of Presentation

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Welding and Repair Technology Center: Extended Storage Collaboration Program Canister Mitigation and Repair Subcommittee – Industry Progress Report

3002013130

Welding and Repair Technology Center: Extended Storage Collaboration Program Canister Mitigation and Repair Subcommittee – Industry Progress Report

3002013130

Technical Update, December 2018

EPRI Project Manager

J. Tatman

All or a portion of the requirements of the EPRI Nuclear Quality Assurance Program apply to this product.

YES NO

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ABSTRACT

Management of used nuclear fuel continues to be a significant issue for nuclear power plants and has been addressed temporarily through fabrication and on-site storage of used nuclear fuel dry cask storage systems. A concern with the current design of dry cask storage systems is the risk of environmental degradation in salt-bearing environments, specifically the initiation of chloride-induced stress corrosion cracking on the stainless steel canisters located within the concrete overpack.

Inspection, mitigation, and repair processes to address chloride-induced stress corrosion cracking concerns for dry cask storage system canisters are being considered and have been developed as a part of the Extended Storage Collaboration Program established by EPRI in 2009. This report identifies collaborative repair and mitigation technology development efforts that are currently underway.

Keywords
Canister
Dry cask storage system
Extended Storage Collaboration Program
Mitigation
Repair
Used nuclear fuel

CONTENTS

ABSTRACT ..... **V**

1 **INTRODUCTION** **1-1**

1.1 Overview of Dry Cask Storage Systems (DCSS) 1-1

1.2 Extended Storage Collaboration Program Background 1-1

1.3 Purpose of ESCP Canister and Mitigation Subcommittee 1-1

2 **CANISTER MITIGATION AND REPAIR: TECHNOLOGY REVIEW** **2-1**

2.1 Applicable Canister Mitigation and Repair Processes 2-1

2.2 Discussion of Selected Processes 2-3

2.2.1 Arc Welding 2-3

2.2.2 Friction Stir Welding and Processing 2-3

2.2.3 Cold Spray 2-5

2.2.4 Low Plasticity Burnishing 2-5

2.2.5 Laser Peening 2-6

3 **CANISTER MITIGATION AND REPAIR TECHNOLOGY DEVELOPMENT: APPROACH AND GUIDELINE** **3-1**

3.1 Three-Phased CM&R Technology Development Approach 3-1

3.1.1 Phase I: Proof-of-Concept Demonstration (Technology Readiness Level: 1–3) 3-2

3.1.2 Phase II: Remote Deployment Demonstration (Technology Readiness Level: 4–5) 3-3

3.1.3 Phase III: Demonstration on Full Canister Mockup (Technology Readiness Level: 6–7) 3-3

3.2 Mockup and Equipment Availability for CM&R Development Efforts 3-3

3.2.1 Mockups Available for Phase I Development Efforts 3-3

3.2.2 Mockups and Equipment Available for Phase II Development Efforts 3-6

3.3 Collaborative Research Opportunities 3-8

4 **SUMMARY** **4-1**

5 **REFERENCES** **5-1**

LIST OF FIGURES

Figure 2-1 FSP repair of laboratory created CISCC in 1" thick stainless steel 304L coupons performed at Pacific Northwest National Laboratory 2-4
Figure 2-2 Schematic of laser peening process 2-6
Figure 3-1 Technology Readiness Levels (TRLs) for evaluating the development state of technologies 3-2
Figure 3-2 Partial ring 304L welded mockups available for canister mitigation and repair technology development efforts 3-4
Figure 3-3 Type 304L stress corrosion cracking plate mockups potentially available for canister mitigation and repair technology development efforts 3-5
Figure 3-4 Schematic of pre-welded v-groove mockup design being fabricated by EPRI for Phase I cold spray development efforts 3-6
Figure 3-5 Full diameter canister mockup with implanted CISCC flaws 3-7
Figure 3-6 DCSS vent pathway mockup used for testing remote delivery inspection systems 3-8

LIST OF TABLES

Table 3-1 Canister mitigation and repair development activities 3-9

1 INTRODUCTION

1.1 Overview of Dry Cask Storage Systems (DCSS)

Long term dry storage systems for used nuclear fuel (UNF) were originally developed with the expectation of being housed permanently within isolated repository sites. Since the establishment of such facilities has yet to come to fruition, this has resulted in dry cask storage systems (DCSS) being fabricated and managed at nearly one hundred independent spent fuel storage installations (ISFSIs) across the United States [1].

A substantial number of ISFSIs have been licensed and developed on-site at various nuclear power plants, some of which are due for license renewal application prior to 2020. One requirement established by the Nuclear Regulatory Commission (NRC) for obtaining ISFSI license renewal is the consideration of degradation mechanisms that could result from DCSS life extension. This requirement by the NRC has raised long-term aging concerns regarding DCSS canisters, specifically the potential for chloride induced stress corrosion cracking (CISCC) in salt-bearing environments.

1.2 Extended Storage Collaboration Program Background

EPRI has conducted various research and development studies aimed towards addressing concerns regarding CISCC in DCSS canisters [2-8]. Due to the scale of this effort and the potential impact on the industry, the importance of obtaining input and collaboration from utilities, national laboratories, universities, and service vendors was recognized by EPRI. A collaborative industry consortium, referred to as the Extended Storage Collaboration Program (ESCP), was therefore initiated by EPRI in 2009 to address long-term fuel storage degradation concerns.

Prior to canister mitigation and repair technology development efforts, one of the primary focuses of ESCP was to develop remotely deployable, in-situ DCSS canister inspection techniques. A publicly available report describing completed and on-going DCSS canister inspection technology development efforts was published by EPRI in 2017 [9]. This task proved to be challenging due to internal clearance restrictions, potential for significant radiation levels, and in-service temperature levels present within the DCSSs. However, through extensive collaboration within the ESCP consortium, remote delivery visual inspection systems have been successfully developed and recently deployed within several UNF-loaded DCSSs.

1.3 Purpose of ESCP Canister and Mitigation Subcommittee

The mission of the ESCP committee is to “provide the technical bases to ensure the continued safe, long-term storage and future transportability of used nuclear fuel”. One of the programs primary objectives is to support the development of industry aging management plans regarding inspection, mitigation, and repair and replacement of UNF storage and transportation systems.

The ability to inspect and potentially detect CISCC in DCSS canisters has resulted in significant interest in preparing mitigation or repair capabilities, should degradation be disclosed during future inspections. As a result, the ESCP consortium created the Canister Mitigation and Repair (CM&R) Subcommittee (SC) in 2016. This CM&R SC has been divided into two task groups to separately manage canister repair and mitigation technology development efforts.

Top repair and mitigation processes for addressing CISCC in DCSS canisters have been down-selected by the CM&R task groups based on specific tabulated criteria. This criteria and the top processes are both detailed in this report. It is important to note that the canister repair and mitigation techniques selected by the task groups are not limited to these top processes and others are likely to be considered based on applicability.

The purpose of the ESCP CM&R SC is not to select, evaluate or recommend a top technique for demonstration, or to identify specific demonstration activities for a given technique, but rather to provide guidance on how industry should approach technique evaluation/demonstration for CISCC mitigation and repair. This guidance is described herein and structured as a three-phased evaluation/demonstration process. The intent of this approach is to provide a consistent and comparable assessment method for industry regarding selection and application of CISCC mitigation and repair techniques.

2
CANISTER MITIGATION AND REPAIR: TECHNOLOGY REVIEW

Two CM&R task groups (TG) have been established that focus on canister mitigation, the Mitigation Task Group (MTG) and canister repair, the Repair Task Group (RTG). These groups operate on a consensus basis and consists of industry subject matter experts (SME) and stakeholders. One purpose of the task groups is to identify and evaluate tools for mitigation and repair of CISCC in DCSS canisters. Tools are defined as techniques, technologies and practices.

Mitigation tools are the various technologies, techniques, practices, etc. that can be used to prevent or mitigate canister degradation that may result from CISCC and other forms of environmental degradation. Mitigation is defined as an action to be taken prior to the discovery of canister defects.

Repair tools are the various technologies, techniques, practices, etc., that may be applied for the repair of DCSS canister defects. Repair is defined as an action to be taken after physical defects are identified.

Members of the MTG and RTG are working to evaluate available technology and determine optimal processes based on specific criteria. Technology evaluation tables are under development as a part of this evaluation and will be provided as a tool for stakeholders to quickly understand various technologies proposed for nuclear repair and mitigation of CISCC. Processes will be ranked using specific criteria, including (but not limited to) previous operating experience, feasibility/accessibility of using the process in-situ or ex-situ, accessibility of the process, effectiveness towards CISCC prevention, inspection impact, heat transfer or structural integrity impact, functionality impact on the canister, and commercial availability. Once finalized, the technology evaluation tables will be included in a future revision to this report.

2.1 Applicable Canister Mitigation and Repair Processes

Based on initial evaluation performed by the CM&R SC, the processes described below have been determined to be the most applicable for canister mitigation and repair. Other techniques may be added and described in future revisions to this report if applicability is recognized.

- **Permanent Repair Techniques**
  - Low temperature friction stir welding and processing (FSW/P) can produce repairs with overmatched mechanical properties and equivalent generalized corrosion performance relative to base metal. It is envisioned that FSW/P repair would be performed with the canister fully or partially removed from the overpack. FSW/P is considered one of the higher rated processes for full structural repair.
  - Arc welding is also considered a highly rated process for full structural repair and is the most recognized and widely-used process in the nuclear power industry. However, the peak temperatures reached during arc welding are substantially higher compared to FSW/P. These higher temperatures could promote sensitization in the base metal heat-

affected zone and result in reduced CISCC resistance on the canister OD surface. For this process to be considered a permanent repair, a surface stress improvement technique (e.g., peening, cold spray) should therefore be used in conjunction with the arc welding repair.

- Cold spray can execute permanent nonstructural repairs within the overpack without detrimental heat input. It is envisioned that high pressure cold spray processes can be qualified as a permeant repair while the canister remains in the overpack. Cold spray is considered the highly rated process for permanent repairs within the overpack.

- **Interim Repair Techniques**

  Various utilities have expressed a desire to execute a rapid repair to ensure immediate confinement and allow for additional review before a more permanent repair is executed. The following processes could be considered for interim repair.

  - Cold spray repair could potentially be executed rapidly without risk of sensitizing base metal, growing cracks or complicating further repair. If a utility discovers a crack, a rapid cold spray repair may be an option to ensure immediate confinement. Once confinement is assured, the utility would then determine if further repair is needed. If so, adding additional thickness to a cold spray repair would likely be trivial. FSW/P could also potentially be performed directly over cold spray repairs if needed.
  - Arc welding may be the most economical method that could be used for confinement of CISCC. Since this is a highly developed process in the nuclear power industry, it is possible that tooling already established for this process could be readily modified to remotely perform interim canister repairs.

- **Mitigation Techniques**

  Three technologies have been ranked as the top processes for achieving mitigation in both new construction and possibly in situ application. As technologies evolve, this ranking may change to reflect methods that are closer to implementation.

  - Cold spray is a process that deposits a metallic layer which can potentially provide enhanced structural properties, corrosion resistance, and compressive residual surface stress properties. Portable equipment is currently available in some commercial applications outside of nuclear.
  - Laser peening introduces a high energy laser beam on the metallic surface to provide high pressure pulses that result in shock waves through the part and apply a layer of residual compressive stress. The laser peening process is commercially available for pre-service canister application.
  - Low Plasticity Burnishing employs a hydrostatic tool that floats a burnishing ball over the surface to impart plastic deformation and develop surface compressive stresses. Application on canisters for pre-service and in-situ conditions need to be developed and demonstrated.

2.2 Discussion of Selected Processes

2.2.1 Arc Welding

Arc welding uses electricity to join materials by melting at the interface. This has been the most widely-used technology for joining and repairing components for nearly a century and is the standard fabrication and repair technology for the nuclear industry and many other industries.

Advantages:

- Commercially available manual and automated equipment
- Robotic solutions for repair within overpack are at a high technology readiness level
- Relevant codes and code cases exist for various arc welding repair processes

Technical Challenges:

- The United States Department of Energy [10] and Nuclear Regulatory Commission [11] have determined that sensitization and residual tensile stresses can be produced from fusion welds on austenitic DCSS canisters. The high substrate heat input generated by the arc welding process can produce localized reduction of chromium near grain boundaries within the stainless steel heat affected zone, resulting in reduced corrosion resistance. The arc welding process therefore promotes a fundamental physics issue of low CISCC resistance within the deposited material and surrounding base metal heat affected zone. However, process controls for arc welding (e.g., low heat input, interpass temperature control) could be implemented to limit or reduce the risk of sensitization.

2.2.2 Friction Stir Welding and Processing

Friction Stir Welding (FSW) is a solid state joining process and effectuated by spinning a tool and plunging it into the workpiece (Figure 2-1). Frictional heating and material deformation enable the formation of a plasticized region below the tool shoulder called the stir zone. The FSW tool and stir zone are traversed across the joint effectuating a weld. Temperatures generated during FSW are typically between 60% and 80% of the absolute melting temperature of the workpiece material. Friction Stir Processing (FSP) is the use of FSW for purposes other than joining such as material property modification and repair.

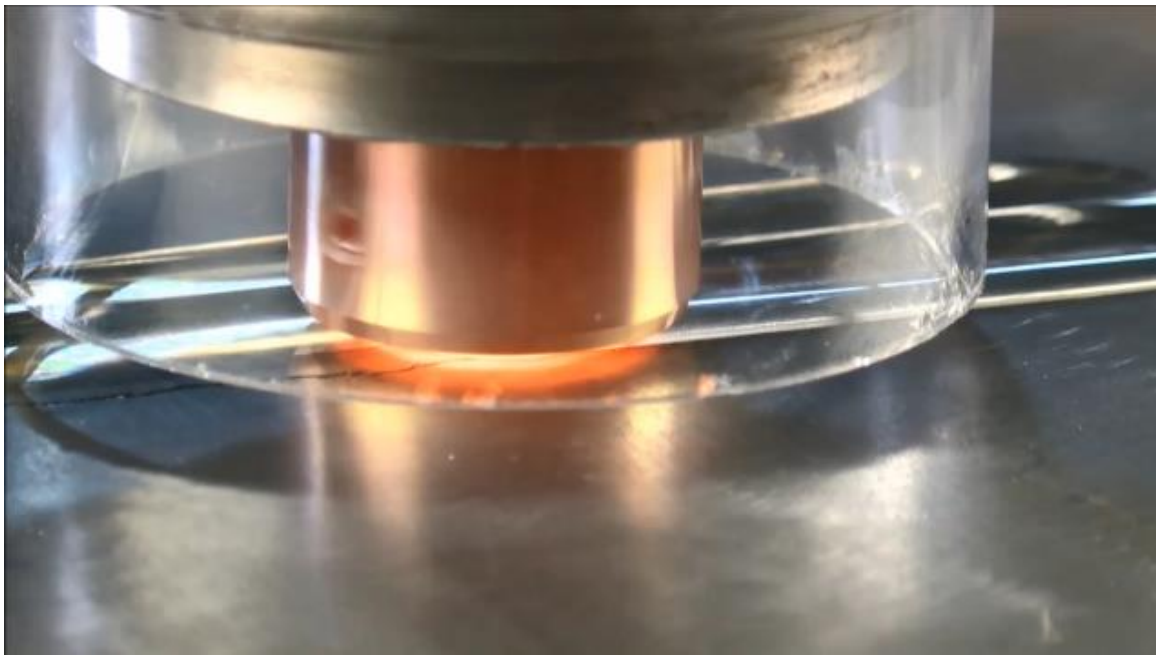


Figure 2-1
FSP repair of laboratory created CISCC in 1" thick stainless steel 304L coupons performed at Pacific Northwest National Laboratory

Advantages:

- Temperature controlled FSP ensures heat input is sufficiently low to avoid reduced corrosion performance in or around the weld
- Mechanical properties are overmatched
- Fully repairs physical defects (cracks, pitting) and microstructural damage (sensitization)
- Green process (no harmful fumes generated, low energy consumption)
- Weld or repairs can be done in a single pass
- Fully automated
- Easily applied in field with overpack removed

Technical Challenges:

- Design of FSW/P equipment to fit between canisters and overpack is a nontrivial engineering challenge and may not be feasible.

FSW/P should be considered for use in new fabrication of canisters and repair of existing canisters. Low heat input and grain refinement result in material with improved mechanical properties and equivalent generalized corrosion performance relative to base metal. Development efforts [7, 8] by PNNL, Fluor and EPRI demonstrated the following: no crack growth during FSP repair operations, cracks are fully consolidated when their depths are less than the stir zone; crack depths larger than the stir zone become stable and can be successfully embedded after application of FSP repair.

2.2.3 Cold Spray

Cold spray is a solid-state deposition process where particles are accelerated to supersonic velocities and impact a substrate. The impact energy is sufficient to plastically deform the material at the interface and produce a mechanical interlock and metallurgical bond [12]. During the process, substrate heating is minimized, dimensional stability is maintained, and unwanted thermal effects (heat affected zone, thermal stresses, dilution layer formation, etc.) are avoided.

Advantages:

- Very low energy input; minimal thermal impact with no detrimental heat affected zone
- Structural properties can potentially be maintained with cold spray deposit repair
- There is no specified limit on deposit thickness, although minimal deposit thickness may be sufficient to arrest cracks and prevent further growth.
- Insignificant formation of porosity is typical in cold spray deposits.
- Cold spray has the potential to produce a high hardness and cold worked microstructure with compressive residual stresses as opposed to the tensile residual stresses associated with arc welding processes. These properties retard crack propagation and improve the corrosion fatigue resistance in a manner similar to peening.
- Oxide formation, alloy decomposition, combustion product entrapment is minimal [12]
- Portable equipment is commercially available

Technical Challenges:

- Design of equipment to effectuate repair with the overpack in place is feasible but challenging.
- Cold spray for full structural repair is an unexplored research topic. Because cold spray is an additive process, building up cold spray material to restore structural integrity to the canister would result in a modification to canister geometry.
- Obtaining high deposit efficiency of the powder can be difficult depending on the application. Methods to capture the remnant powder during cold spray deposition may be required.

2.2.4 Low Plasticity Burnishing

Low plasticity burnishing (LPB®) is a patented surface stress improvement process that has the potential to provide protection against CISCC. This process was developed by Lambda Technologies Group and has seen use in applications for over two decades. The LBP® process involves the use of a “floating” ball to continuously burnish the material surface and produce compressive residual stresses. According to Lambda Technologies Group [13], the LPB® process induces minimal plastic deformation and can produce surface residual compressive stresses at depths ranging between one millimeter and over one centimeter. Due to the simplicity of the tooling used for this process, it is also likely that this LPB® process could be configured to fit within the tight DCSS constraints.

2.2.5 Laser Peening

Peening is a process in which a compressive force is applied to the surface of a material with the goal of slightly deforming the surface to reduce tensile residual stresses. Traditionally this is done by firing small metal or glass beads at the surface of the part at high velocity causing small deformations. This same process can be performed using lasers, in which the part is coated with a sacrificial absorbing ablative material (such as paint or tape) which reduces the potential for thermal damage on the surface from the laser (Figure 2-2). The sacrificial layer is vaporized by the laser, producing a high-pressure plasma that is concealed by a film of water to induce plastic deformation on the material surface, resulting in the formation of compressive residual stresses. There are three primary advantages of laser peening: formation of compressive residual stresses at substantial depths, higher material hardness, and refined grain size on the surface. Further, recommended parameters for laser peening on metallics have been issued by SAE in Specification AMS-2546.

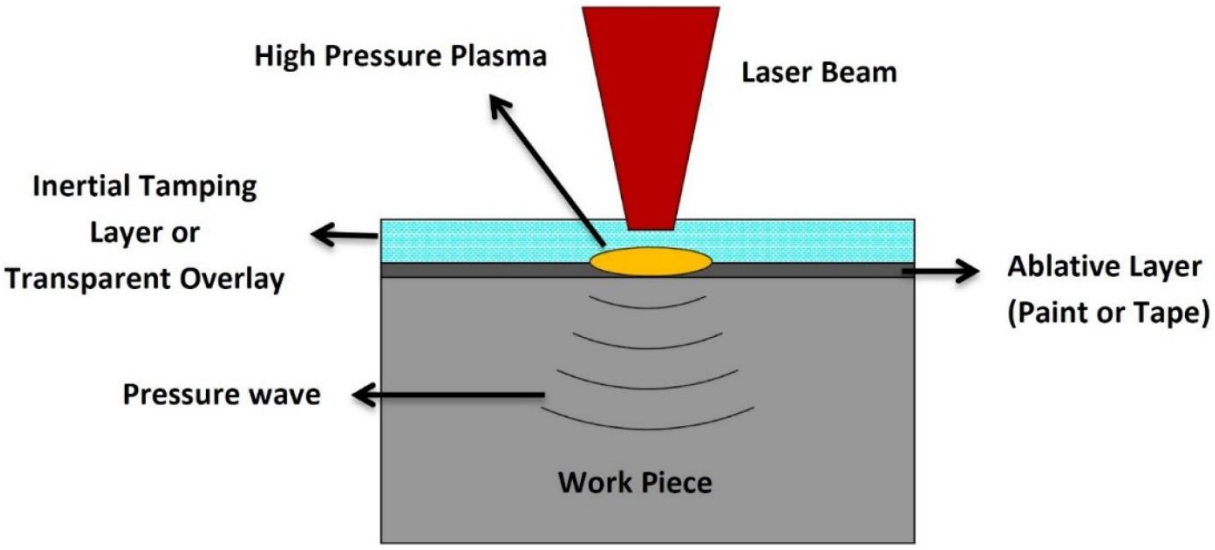


Figure 2-2
Schematic of laser peening process

In addition, surface stress improvement processes such as laser peening have received support from industry and regulatory authorities for use on dissimilar metal welds and reactor head penetrations through EPRI sponsored research. EPRI Report, 3002008083 (MRP-267R2) [14] provides the technical basis for primary water stress corrosion cracking (PWSCC) mitigation by surface stress improvement (including laser peening). Inspection relief for areas mitigated by surface stress improvement has been granted by the NRC through application specific relief requests. These requests have cited EPRI Report 3002009241 (MRP-335R3) [15] which provides the test data and analysis results demonstrating that the peening process meets the performance criteria provided in the report. Also, Holtec has successfully applied the laser peening process prior to loading as a mitigating technique for canisters [16].

3
CANISTER MITIGATION AND REPAIR TECHNOLOGY DEVELOPMENT: APPROACH AND GUIDELINE

3.1 Three-Phased CM&R Technology Development Approach

A process demonstration plan is currently under development by the ESCP CM&R SC and is divided into three phases based on mitigation/repair technology maturity. Guidance is provided herein for each of the phases and contains information concerning the various activities to be considered when preparing a technique for demonstration. The intention of this guidance is to provide a baseline testing approach that ensures an appropriate, comprehensive, and comparable technique assessment. Due to considerable variances in the techniques that may be considered, specific mockup/equipment designs and CISCC evaluation tests will be omitted from this guidance. Rather, the guidance will be structured in a manner that, if properly followed, will help industry develop technology/techniques throughout the Technology Readiness Level maturation cycle (Figure 3-1).

This three-phased approach addresses demonstration activities applicable to both in-service mitigation and repair techniques. Demonstration of mitigation techniques for new fabrication is not explicitly addressed, but the approach will be similar with reference to remote delivery equipment development being omitted.

A descriptive overview of each phase along with specific information regarding the baseline activities to consider for preparing technologies for canister mitigation or repair demonstration is provided below. Reference to Technology Readiness Levels (TRL) provides a common understanding of the maturity status of the technique / technology associated with each phase. Note that compliance to a given code or standard is not inferred.

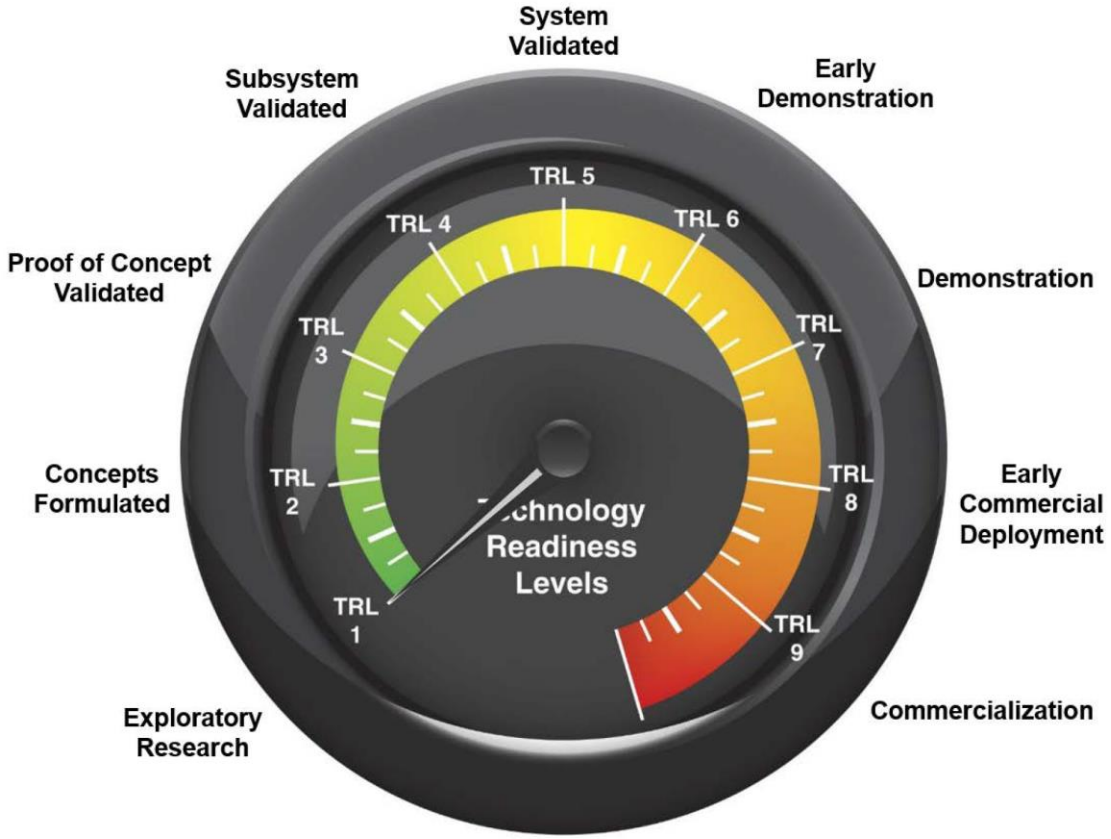


Figure 3-1
Technology Readiness Levels (TRLs) for evaluating the development state of technologies

3.1.1 Phase I: Proof-of-Concept Demonstration (Technology Readiness Level: 1–3)

Phase I demonstration focusses on proof-of-concept activities for mitigation/repair techniques. Included are the following activities:

- Review of studies and research that demonstrate the basic principles of the technique
- Demonstration of basic technology principles, including laboratory/bench scale efforts
- Laboratory testing to validate technique performance and function
- Evaluation of Phase I results to determine acceptability for Phase II demonstration

3.1.2 Phase II: Remote Deployment Demonstration (Technology Readiness Level: 4–5)

Phase II demonstration covers technology and system configuration that is near prototypical. In addition, technology and equipment associated with remote deployment in a simulated environment will be demonstrated. Included are the following activities:

- High-fidelity testing to validate technique performance and function, along with demonstration of remote delivery equipment and components
- Demonstration of integrated technique and delivery system in a remote, simulated environment – laboratory-scale mockup and conditions relevant to the reference application
- Evaluation of Phase II results to determine acceptability for Phase III demonstration

3.1.3 Phase III: Demonstration on Full Canister Mockup (Technology Readiness Level: 6–7)

Phase III includes actual system (technique and remote delivery equipment), full-scale integrated demonstration. Demonstration would likely be performed at the end-user facility on actual DCSS canister.

Following successful Phase III demonstration, the technique and associated remote delivery system will be considered ready for commercial deployment.

3.2 Mockup and Equipment Availability for CM&R Development Efforts

A variety of mockups have been obtained and are being developed by EPRI through the ESCP program which could be used for Phase I and II CM&R technology development efforts. Further, it has been proposed that the remote robotic inspection delivery systems developed by the ESCP NDE subgroup could potentially be modified for CM&R techniques during Phase II evaluation.

Industry, universities, and national laboratories are encouraged to request the following mockups available at EPRI for CM&R technology development efforts.

3.2.1 Mockups Available for Phase I Development Efforts

Three type 304L stainless steel partial ring mockups containing ER308L stainless steel welds (Figure 3-2) are available for Phase I technology development. These mockups contain a radius comparable to a typical DCSS canister design and could potentially be used for CM&R process parameter development studies.



Figure 3-2
Partial ring 304L welded mockups available for canister mitigation and repair technology development efforts

In addition to the three partial ring mockups described above, five type 304L stainless steel plates with SCC flaws (Figure 3-3) are currently being fabricated by EPRI. The potential purpose of these mockups could be to evaluate the ability of CM&R techniques to seal cracks representative of CISCC.

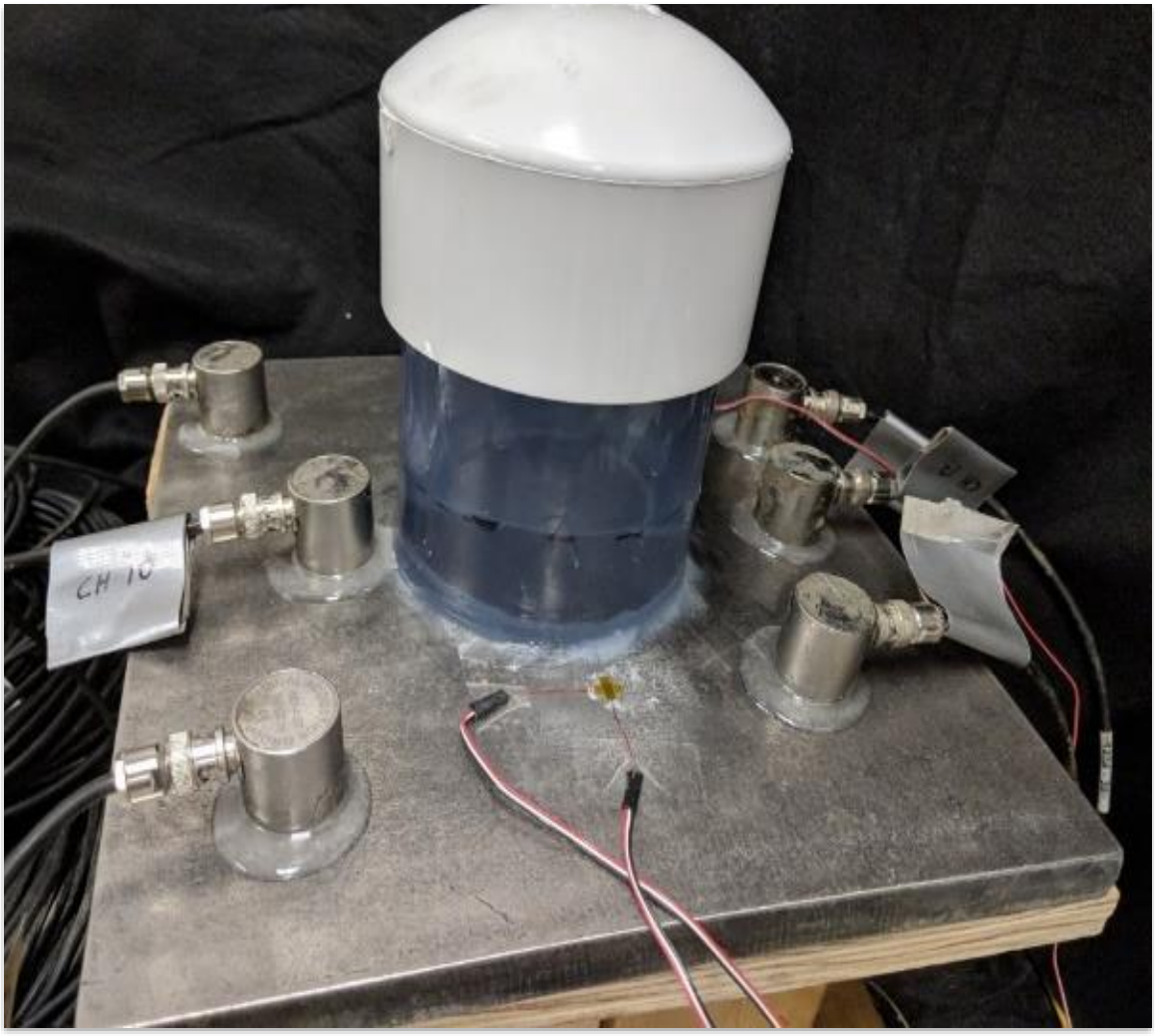


Figure 3-3
Type 304L stress corrosion cracking plate mockups potentially available for canister mitigation and repair technology development efforts

As described in the previous section of this report, the cold spray technique was selected by the ESCP CM&R SC as a top process that should be evaluated for both DCSS canister mitigation and repair. EPRI is in the process of fabricating mockups to prepare for Phase I evaluation of this technique. A schematic of the mockup design in the pre-welded stage is provided in Figure 3-4.

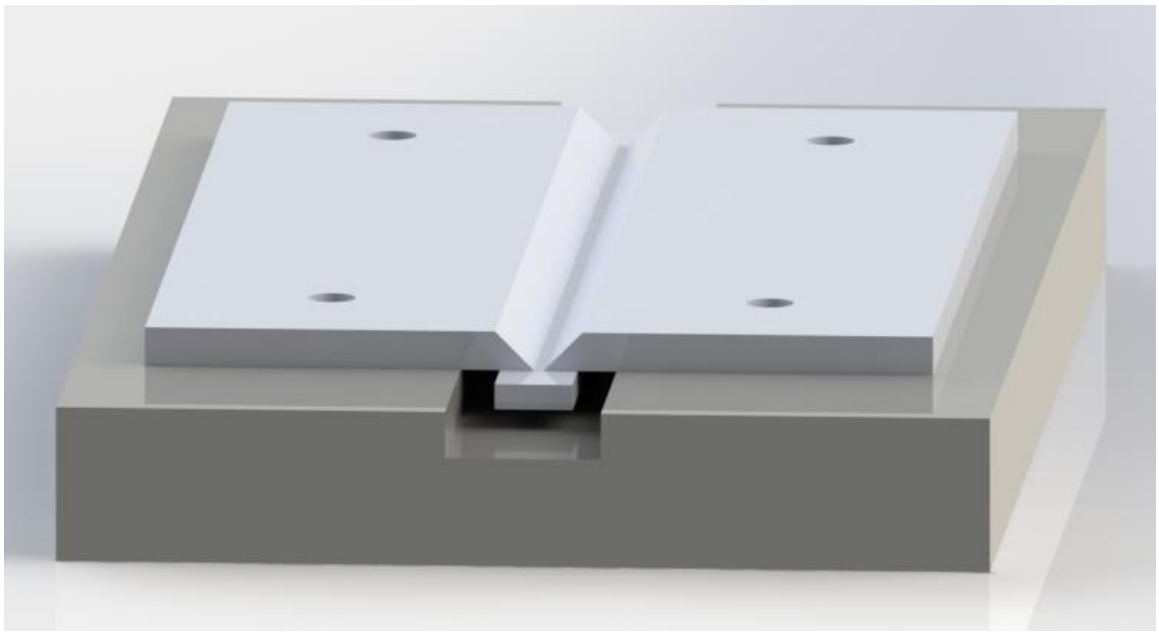


Figure 3-4
Schematic of pre-welded v-groove mockup design being fabricated by EPRI for Phase I cold spray development efforts

Several of these cold spray mockups are being fabricated by EPRI and each will contain an ER308L groove weld deposited using the gas tungsten arc welding (GTAW) process. The base material will be 1.27 centimeter (0.5 inch) thick type 304L stainless steel plates restrained on a carbon steel strong back using a properly designed bolting pattern. The configuration will also include a 304L backing strip tack welded to the bottom of the beveled 304L plates. Following v-groove weld completion, cold spray build-ups will be deposited over approximately half of the ER308L welded region and the surrounding heat-affected zone.

The purpose of these cold spray mockups will be to support various studies planned in 2019 by participants in the ESCP group. The ESCP CM&R committee is also aware of recent Nuclear Energy University Program (NEUP) awards granted to several universities by the DOE, some of which include CM&R cold spray development and corrosion testing efforts. The cold spray mockups being fabricated by EPRI could potentially be used to support these DOE funded studies.

3.2.2 Mockups and Equipment Available for Phase II Development Efforts

Appropriate Phase II evaluation will require test setups representative of DCSS vent pathways, robotic carriage systems capable of remotely delivering the techniques and full diameter canister mockups.

A full-diameter stainless steel canister mockup has recently been donated to EPRI from Sandia National Laboratory (Figure 3-5). This mockup was originally used for non-destructive testing evaluation and contains a stainless steel weld along its vertical length. Further, Pacific Northwest National Laboratory (PNNL) sponsored the implantation of flaws within this mockup that are representative of CISCC. These features make the mockup suitable for future Phase II CM&R technology development efforts.



Figure 3-5
Full diameter canister mockup with implanted CISCC flaws

VRC Metal Systems, in collaboration with South Dakota School of Mines and Technology (SDSM&T) and Robotic Technologies of Tennessee (RTT), received a Small Business Innovation Research (SBIR) award from the DOE in 2017 to develop in-situ CISCC mitigation and repair technology for canisters [17]. Specifically, this effort included the development of robotic cold spray technology for DCSS CM&R. Although initial results were promising, additional Phase I and II development and testing efforts are needed to fully evaluate the technique and equipment. Continuation of this work is recommended by the ESCP CM&R SC for Phase II evaluation of the cold spray technique.

In addition to the SBIR project, it has been proposed that the remote carriage systems already developed and deployed for in-situ canister inspection could potentially be adapted to support mitigation and repair techniques. Modification of these inspection delivery systems should therefore be considered when performing Phase II evaluation of CM&R techniques.

Canister vent pathway mockups have been previously developed for testing inspection delivery systems. An example of one of these vent pathway mockups is provided in Figure 3-6. These mockups could potentially be sought after for use during future Phase II CM&R technology development efforts.



Figure 3-6
DCSS vent pathway mockup used for testing remote delivery inspection systems

3.3 Collaborative Research Opportunities

Multiple activities are underway for development of CM&R technology. To monitor these activities, the ESCP CM&R SC has created an unofficial list (Table 3-1) containing the project title, affiliation, and a short summary of each activity of which the committee is aware. The ESCP CM&R SC will continue to monitor these activities and intends to establish communication with the technical contacts to leverage potential future collaboration opportunities. This list is being maintained by the ESCP CM&R SC for information and tracking purposes. For detailed information or the official status of an activity, contact the affiliated organization.

Table 3-1
Canister mitigation and repair development activities

Project/Activity	Affiliation	Short Summary
In Situ Stress Corrosion Crack Repair and Prevention for Nuclear Waste Canisters [17]	VRC Metal Systems	Development of high velocity cold spray for repair/mitigation of canisters and a robotic delivery system for in-situ applications within casks.
Mobile Examination & Remediation Facility (MERF) [18]	Oak Ridge National Laboratory	Design of a mobile facility to perform repair/mitigation activities of casks - requires removing casks from overpack - no demonstration or equipment development.
DCSS Repair & Mitigation via FSW	Pacific Northwest National Laboratory	Development of low energy temperature-controlled FSW/P to repair/mitigate CISCC and perform closure welds. Development of cold spray to repair/mitigate CISCC.
PNNL NRC Mockup - Inspection Focus - horizontal canister design		Development and validation of NDE techniques for DCSS canisters
CISCC Analysis and Competitive Benchmarking		High fidelity CISCC initiation and growth rate testing.
Stainless Canister In-Situ Robotic Flaw Mitigation/Repair Using Weld Overlay: Feasibility Assessment	Diakont [19]	Feasibility assessment of in-situ application of weld overlays for canister mitigation/repair.
Spent Fuel Canister Residual Stress Analysis/Chemistry of SCC/CISCC Model Development	Sandia National Laboratory [20]	Many projects related to spent fuel canister management and understanding contributing factors such weld residual stress.
Peening during fabrication for CISCC Mitigation [16]	Holtec	Laser peening of Holtec MPC design to support San Onofre.
Repair and Mitigation of Chloride-Induced Pitting and Chloride-Induced Stress Corrosion Cracking in Used Nuclear Fuel Dry Cask Canister Materials [21]	Ohio State University	Several processes are being evaluated in this project for canister mitigation and repair, including cold spray, low temperature friction stir welding, vaporizing foil actuator welding and soldering. After initial evaluation of the four processes, the top two technologies will be selected for more comprehensive development.

Table 3-1 (continued)
Canister mitigation and repair development activities

Project/Activity	Affiliation	Short Summary
Cold Spray Repair & Mitigation of Stress Corrosion Cracks in Spent Nuclear Fuel Dry Storage Canisters [21]	Purdue University	Cold spray process evaluation, parameter optimization and remote delivery system development for canister CISCC mitigation and repair is the primary focus of this study.
Development of Repair and Mitigation Methods for Enhancing Stress Corrosion Cracking Resistance of Austenitic Stainless Steel Spent Nuclear Fuel Canisters [21]	University of Cincinnati	In this study, laser-assisted cold spray and additive friction stir processes are being evaluated and developed for CISCC mitigation and repair of canisters. This project will also investigate the benefit of applying surface stress improvement techniques in conjunction with the mitigation and repair processes. Surface stress improvement processes to be evaluated include laser shock peening and ultrasonic nanocrystal surface modification.
Friction Stir Based Repair Welding of Dry Storage Canisters and Mitigation Strategies: Effect of Engineered Barrier Layer on Environmental Degradation [21]	University of Idaho	Friction stir process development for CISCC mitigation and repair is the primary focus of this study.
Low-Force Solid-State Technologies for Mitigation of Stress Corrosion Cracking in Dry Storage Canisters [21]	University of Wisconsin	Two processes will be evaluated and developed in this study for canister mitigation and repair. These include additive friction stir welding and the cold spray deposition processes.
Geopolymer Repair Development for DCSS [22]	Électricité de France (EDF)	Develop geopolymeric sealant ribbon with good adhesion characteristics, determine service life, develop automated application approach.

4
SUMMARY

Development of in-situ DCSS canister inspection, mitigation and repair technology has become increasingly important due to the potential for CISCC occurrence in salt bearing environments. Visual inspection techniques have been successfully developed by the ESCP committee and recently deployed on several UNF-loaded DCSS canisters. This has driven the need for development of CM&R techniques, as future inspections could disclose unacceptable degradation that requires action for continued service.

The ESCP CM&R SC was developed to manage the development of mitigation and repair technology for DCSS canisters. Common mitigation and repair processes were ranked by the CM&R SC using a comprehensive evaluation criterion and the top processes selected include arc welding, cold spray, friction stir welding, peening, and low plasticity burnishing. A three-phased approach for developing these top processes has been proposed in this report and mockups are available at EPRI for future development efforts. Further, a technology demonstration plan is under development by the CM&R SC that will outline the steps necessary to prepare canister mitigation and repair processes for commercial deployment.

Various activities both internal and external to the ESCP committee are either planned or underway to develop mitigation and repair techniques for DCSS canisters. These have been summarized in this report and include studies being performed by national laboratories, universities, and service vendors.

The significant number of studies underway highlight the need for the CM&R SC to develop an organized technology development plan that will ensure consistent and comprehensive evaluation methods are used to adequately prepare CM&R techniques for commercial deployment. The CM&R SC intends to use the ESCP consortium as a forum to bring together the principal investigators of these studies. The primary intent will be to promote collaboration and steer the efforts in an organized manner that provides the greatest benefit to industry.

5
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 24, 2019

EA-18-151

Dr. K. P. Singh
President and CEO
Holtec International
Krishna P. Singh Technology Campus
1 Holtec Boulevard
Camden, NJ 08104

SUBJECT: HOLTEC INTERNATIONAL – NOTICE OF VIOLATION; U.S. NUCLEAR REGULATORY COMMISSION INSPECTION REPORT NO. 07201014/2018-201 DIVISION OF SPENT FUEL MANAGEMENT

Dear Dr. Singh:

This letter refers to the U.S. Nuclear Regulatory Commission (NRC) announced routine inspection at your Holtec International (Holtec) corporate office in Camden, New Jersey from May 14-18, 2018, with continued in-office review through November 26, 2018. The purpose of the inspection was to assess the adequacy of Holtec's activities with regard to the design of spent fuel storage casks with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-level Radioactive Waste, and Reactor-related Greater Than Class C Waste." Based on the information developed during the inspection, two apparent violations were identified. The circumstances surrounding these apparent violations, the significance of the issues, and the need for lasting and effective corrective actions were discussed with Mr. Mark Soler of your staff during an exit meeting on November 26, 2018. Details regarding the apparent violations were provided in NRC Inspection Report No. 07201014/2018-201, dated November 29, 2018. The inspection report can be found in the NRC's Agencywide Documents Access and Management System (ADAMS) at Accession Number ML18306A853. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

The letter transmitting the inspection report also provided you with the opportunity to address the apparent violations identified in the report by either: (1) attending a Pre-decisional Enforcement Conference (PEC), or (2) participating in an Alternative Dispute Resolution session before we made our final enforcement decision. In a letter dated December 3, 2018, (ADAMS Accession Number ML18341A126) you requested a pre-decisional enforcement conference.

A PEC was convened at the NRC Headquarters on January 9, 2019, with you and members of your staff to discuss the violations. Subsequent to the PEC, you submitted additional information for the NRC to consider as we proceeded with our enforcement decision process.

The NRC has determined that two violations of regulatory requirements occurred. This determination was based on information developed during the NRC inspection, information you provided in your responses to the inspection report, and information you provided during and after the PEC. The violations are cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding them are described in detail in the subject inspection report (ML18306A853). These violations involved: (1) failure to establish adequate design control measures as a part of the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the functions of the structures, systems, and components which are important to safety as required by 10 CFR 72.146(a), and (2) failure to perform 10 CFR 72.48 evaluations prior to implementing proposed changes and failure to obtain certificate of compliance (CoC) amendments pursuant to 10 CFR 72.244 as required by 10 CFR 72.48.

The failure to establish adequate design control measures and obtain NRC approval prior to modifying multi-purpose canisters (MPC) with four-inch stainless steel stand-off pins, was deemed potentially safety significant. Holtec's design review process for the change did not adequately consider all potential impacts that could adversely affect the safety-related functions of the MPC shims. Impacts not adequately considered included, but are not limited to: MPC handling and manufacturing processes to include peening, lateral loads based on gaps within the MPCs, and conditions adverse to quality identified when personnel discovered defects with the shim stand-off pins during installation. The stand-off pins are essential to the function of the fuel basket to maintain support and ensure that the shims remain in place to allow helium to adequately circulate around the fuel assemblies within the canister.

The MPCs with the shim stand-off pins were not loaded to the full design basis heat load; however, the potential loss of multiple shim stand-off pins in an MPC loaded at the design-basis heat load configuration could have compromised the heat transfer characteristics of the MPC. This could have resulted in an increase in the peak cladding temperature beyond the allowable limit and potentially damage the fuel cladding. Holtec performed a thermal analysis to assess the consequences of the failure of multiple stand-off pins within a MPC and determined that, based on the assumptions used, all predicted results would remain below the described limits in the final safety analysis report with acceptable margin.

NRC staff reviewed the Holtec analysis and concluded that the heat transfer characteristics of the MPCs were adequate and that loaded MPCs would continue to be in a safe condition during the entire licensed period of storage as described in the respective CoC. As a result of this review, the NRC determined that Holtec's violation of 10 CFR 72.48 and 10 CFR 72.146 did not result in an actual significant safety concern. However, the NRC considers Violation 1 to be of moderate safety significance because Holtec's inadequate design control measures did not adequately assess a potentially credible accident and exposure scenario that had the potential for a significant consequence. The failure of multiple stand-off pins in an MPC could have resulted in inadequate heat transfer and the exceedance of peak clad temperature limits. Therefore, this violation has been categorized in accordance with the NRC Enforcement Policy at Severity Level III, in part, because the design change was outside design specifications to the extent that a detailed evaluation was required to determine its operability. The current Enforcement Policy is included on the NRC's website at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html.

Holtec has not been the subject of escalated traditional enforcement action within the last two years or two previous inspections. The NRC did not identify any aggravating circumstances through the course of its enforcement decision process. For mitigating circumstances, the NRC

considered whether credit was warranted for *Corrective Action* in accordance with the civil penalty assessment process in Section 2.3.4 of the Enforcement Policy.

The NRC determined *Corrective Action* credit was warranted due to corrective actions initiated by Holtec. The following are the descriptions of short-term and long-term corrective actions taken. Short-term corrective actions and actions to prevent recurrence for violation 1 included: (1) a root cause evaluation; (2) an analysis of licensees loaded units to ensure they are in safe condition; (3) an analysis of basket shim stand-off for seismic and impact loading; (3) inspection of all non-loaded units to identify necessary actions on a case-by-case basis; (4) elimination of the shim stand-off design from MPC's licensing and fabrication drawings; and (5) notification to customers that have delivered or loaded systems.

Long-term corrective actions included: (1) conducting a lessons learned assessment to cover evaluation of design change from a manufacturing and licensing/analysis standpoint and to address issues within the corrective action program; (2) revising engineering change orders (ECO) and drawing review checklists to include questions on impacts to components during fabrication activities; (3) development of written instructions for process change risk evaluations; (4) development of on-the-job training to include shop tours and review of standard manufacturing processes; (5) evaluation of corrective actions initiated in 2018 for design changes that may not have been appropriately evaluated; (6) development of an ECO surveillance process for technical discipline managers to assess whether design changes were appropriately evaluated; (7) training shop personnel on reviewing travelers and other written instructions prior to performing work; (8) training shop personnel on issues identified with installation of the shims; and (9) evaluating the design change process within technical disciplines to determine areas for improvement.

A base civil penalty in the amount of $36,250 was considered for this Severity Level III violation. However, the staff determined, in accordance with the Enforcement Policy, that a civil penalty for Violation 1 was not warranted. This determination is in recognition of no aggravating circumstances, Holtec's prompt and comprehensive correction of the violation, and the absence of recent escalated enforcement action. This enforcement action may be used in the evaluation of any future significant violations, which could result in a civil penalty. In addition, issuance of the Severity Level III violation constitutes escalated enforcement action that may subject you to increased inspection effort. The NRC includes significant enforcement actions on its Web site at http://www.nrc.gov/reading-rm/doc-collections/enforcement/actions/.

The NRC determined that Violation 2 is a Severity Level IV violation with three examples of failing to follow NRC's requirement to adequately perform a 10 CFR 72.48 evaluation prior to implementing proposed changes and failing to obtain CoC amendments pursuant to 10 CFR 72.244. The violation is described in the subject inspection report and was evaluated in accordance with the NRC Enforcement Policy and resulted in conditions as having very low safety significance. The NRC has determined that escalated enforcement was not warranted for this violation. The violation is cited in the enclosed Notice of Violation (Notice). There is no civil penalty associated with a Severity Level IV violation.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. If you have additional information that you believe the NRC should consider, you may provide it in your response to the Notice. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements. In addition, we will follow up your corrective actions during a future NRC inspection.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if any, will be made available electronically for public inspection in the NRC Public Document Room and in ADAMS, accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such information, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). The NRC also includes significant enforcement actions on its Web site at http://www.nrc.gov/reading-rm/doc-collections/enforcememt/actions/).

Should you have any questions, please contact Mr. Christian Araguas at (301) 415-7210 or e-mail (Christian.Araguas@nrc.gov).

Sincerely,

George Wilson, Director
Office of Enforcement

Docket Nos. 72-1014, 72-1040,
and 72-1032

Enclosure:
Notice of Violation

NOTICE OF VIOLATION

Holtec International
Camden NJ

Docket Nos. 72-1014, 72-1040, 72-1032
EA-18-151

Based on the results of an U.S. Nuclear Regulatory Commission (NRC) inspection conducted at Holtec International (hereafter referred to as Holtec), on May 14, 2018, through July 19, 2018, a team of inspectors identified violations of NRC requirements. In accordance with the NRC Enforcement Policy dated May 15, 2018, the violations are listed below:

Violation 1

10 CFR 72.146(a), "Design control," requires, in part, that measures must be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the functions of the structures, systems, and components which are important to safety.

Contrary to the above, Holtec failed to establish adequate design control measures as a part of the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the functions of the structures, systems, and components which are important to safety. Specifically, on or after August of 2016, Holtec failed to establish adequate design control measures as a part of the selection and review for suitability of application for alternative four-inch stainless steel stand-off pins that were essential to the function of the fuel basket to maintain support and ensure that the shims stay upright to allow airflow to the fuel assemblies within the MPC.

This is a Severity Level III violation (NRC Enforcement Policy, Section 6.3.c.1 (b))

Violation 2

A. 10 CFR 72.48(c)(1)(ii)(A), requires, in part, that a certificate holder may make changes in the facility or spent fuel storage cask design as described in the final safety analysis report (FSAR) (as updated), make changes in the procedures as described in the FSAR (as updated), and conduct tests or experiments not described in the FSAR (as updated), without obtaining a CoC amendment submitted by the certificate holder pursuant to § 72.244 (for general licensees and certificate holders) if a change to the technical specifications incorporated in the specific license is not required.

Contrary to the above, as of July 19, 2018, the certificate holder (Holtec) did not obtain a CoC amendment pursuant to § 72.244 for a storage cask design as described in the FSAR despite the fact that the new HI-TRAC VW, Version V design, required a change to the technical specification (TS) incorporated in the CoC. Specifically, Holtec made a change to the HI-TRAC VW design that required new operator actions with new dose rates that affected the FSAR design function and specifications. This change substituted a manual action for an automatic action for performing an FSAR described design function, which would require prior NRC

approval because it would result in more than a minimal increase in the likelihood of occurrence of a malfunction and a change to the TS.

B. 10 CFR 72.48(c)(2)(v), "Changes, tests, and experiments," requires, in part, that a certificate holder shall obtain a CoC amendment pursuant to 10 CFR 72.244 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would create a possibility for a malfunction of a different result than any previously evaluated in the FSAR (as updated).

   Contrary to the above, as of July 19, 2018, the certificate holder (Holtec) did not obtain a CoC amendment when implementing a proposed change that would create a possibility for a malfunction of a different result than any previously evaluated in the FSAR. Specifically, personnel were unable to remove a damaged lift cleat threaded stud from a lifting hole which resulted in three instead of four functional MPC lifting points. Holtec's Hi-Storm 100 FSAR analyzed lifting a fully loaded cask with four lifting lugs.

C. 10 CFR 72.48(d)(1) requires, in part, that the licensee and certificate holder shall maintain records of changes in the facility or spent fuel storage cask design, of changes in procedures, and tests and experiments made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change does not require a CoC amendment pursuant to paragraph (c)(2) of this section.

   Contrary to the above, as of July 19, 2018, the certificate holder (Holtec) failed to maintain records of changes that included a written evaluation that provided the bases for the determination that the change does not require a CoC amendment pursuant to 10 CFR 72.48(c)(2). Specifically, Holtec failed to perform a written evaluation to demonstrate that a design change for multi-purpose canister stainless steel standoff pins did not require a CoC amendment. Holtec completed a 10 CFR 72.48 screening and incorrectly determined that a written evaluation was not needed.

This is Severity Level IV violation (NRC Enforcement Policy, Section 6.1.d.2).

Pursuant to the provisions of 10 CFR 2.201, Holtec is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to Damaris Marcano, Acting Chief, Inspections and Operations Branch, Division of Spent Fuel Management, Office of Nuclear Material Safety and Safeguards, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level; (2) the corrective steps that have been taken and the results achieved; (3) the corrective steps that will be taken to avoid further violations; (4) your plan and schedule for completing short and long term corrective actions and (5) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or

revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/readingrm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21. If Classified Information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR Part 95. In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days of receipt.

Dated this 24th day of April 2019.

HOLTEC INTERNATIONAL

Krishna P. Singh Technology Campus, 1 Holtec Blvd., Camden, NJ
Telephone (856) 797-0900
Fax (856) 797-0909

April 26, 2019

Mr. Michael C. Layton, Director
Division of Spent Fuel Management
Office of Nuclear Material Safety
and Safeguards
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

Via Email:
Michael.Layton@nrc.gov

Subject: Holtec International – Notice of Violation; U.S. Nuclear Regulatory Commission Inspection Report No. 07201014/2018-201
Division of Spent Fuel Management

Reference: Holtec Project #5014

Dear Director Layton:

We thank you for your letter of April 24, 2019 on the above-referenced subject matter. We accept NRC's characterization of the violations and of our investigations and evaluations that followed the discovery of the shim stand-off issue, including the root cause evaluation and corrective action efforts. We appreciate the credit granted to us for the proactive corrective action implementations made by us and represent to you that all remedial actions arising from the shim stand-off episode documented in our QI#2471 have been fully integrated into the infrastructure of our QA program. A formal response as requested in your letter will follow within 30 days.

We would welcome NRC's validation of our QA upgrades to ensure unconditional compliance through future scheduled or unscheduled inspections. We are certain that the NRC's inspection team will confirm that we are not only in comprehensive compliance with our representations made to the NRC regarding the corrective actions arising from this issue but have made many other enhancements as a part of our continuous improvement program to insure that our QA program remains state-of-art. In particular, our recent emphasis has been in erecting more effective barriers against human performance errors (of the sort that led to the shim stand-off

HOLTEC INTERNATIONAL

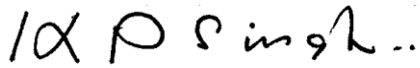
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issue) throughout the life cycle of the product - from design conceptualization - to site commissioning.

We are heartened to note that NRC agreed with our technical position that the presently deployed MPCs equipped with the shim stand-offs meet all applicable regulatory criteria.

Again, we thank you for your balanced evaluation and fact-based conclusions to close out this subject matter.

Sincerely,



Dr. Kris Singh, President & CEO
Holtec International

cc: Dr. Stefan Anton
Kim Manzione
Mark Soler
EXCOM

Document No. 5014872

April 30, 2019

The Honorable Lamar Alexander
Chairman, U.S. Senate Subcommittee on Energy & Water Appropriations
455 Dirksen Senate Office Building
Washington, DC 20510

The Honorable Dianne Feinstein
Ranking Member, U.S. Senate Subcommittee on Energy & Water Appropriations
331 Hart Senate Office Building
Washington, DC 20510

The Honorable Marcy Kaptur
Chair, U.S. House Subcommittee on Energy & Water Appropriations
2186 Rayburn House Office Building
Washington, DC 20515

The Honorable Mike Simpson
Ranking Member, U.S. House Subcommittee on Energy & Water Appropriations
2084 Rayburn House Office Building
Washington, DC 20515

Dear Senate and House Energy & Water Appropriations Leaders:

The undersigned organizations seek your leadership to appropriate funds in FY 2020 for the following critical elements of an effective nuclear waste management program:

- **Completion of the Yucca Mountain Licensing Review.** Consistent with the Nuclear Waste Policy Act (NWPA), the U.S. Nuclear Regulatory Commission (NRC) and the U.S. Department of Energy (DOE) need funding as requested to proceed with the adjudicatory process for evaluating scientific evidence provided by experts from DOE, the State of Nevada, and other parties and determining whether the license application for the proposed Yucca Mountain repository should be granted.
- **Implementation of Pilot Consolidated Interim Storage with Priority for Shutdown Reactor Fuel.** Congress should also facilitate a pilot consolidated interim storage (CIS) facility for spent nuclear fuel (SNF) and Greater-Than-Class-C (GTCC) waste stranded at shutdown nuclear sites to (i) enable the government to begin meeting its obligations sooner; (ii) allow for other uses of decommissioned reactor sites; and (iii) implement the country's SNF transportation infrastructure. So that this pilot can be made available in a timely manner, DOE should be instructed to work with private initiatives that are already pursuing licenses for such a facility to the maximum extent possible.
- **Preparation for SNF Transportation.** Because development of SNF transportation infrastructure will be necessary regardless of destination, we support funding for DOE to (i) test, certify, and procure rail cars, as well as licensed transportation casks and components in a manner that utilizes the private sector to the maximum extent practicable as required by the NWPA; and (ii) increase financial and technical assistance to tribal, state, and local governments for transportation and related emergency preparedness activities that will enhance public confidence in the safety of SNF transportation.

We ask that your respective Energy and Water Appropriations Subcommittees include these three items in your upcoming markups. It is imperative that Congress provide direction and funding this session to advance an integrated national nuclear waste program that includes completing the Yucca Mountain license application review and establishing a pilot CIS facility with priority for shutdown reactor fuel. In addition, we urge you to restructure the funding and spending mechanisms for the Nuclear Waste Fund (NWF) to provide the necessary certainty to implement the program over multiple decades. This should include sustainable access to the NWF

balance and all accumulated interest while maintaining effective Congressional oversight. Together, these actions would send a clear message that you are committed to resolving the nuclear waste stalemate and ensuring that the federal government carries out its statutory and contractual responsibility to manage and dispose of SNF.

Urgent action is needed. NWF receipts total nearly $50 billion, including fees collected from U.S. electric consumers per the NWPA and interest accumulating at a rate of approximately $1.5 billion per year. Electric consumers have met their obligation and received nothing in return. In addition, the impasse over nuclear waste is costing all U.S. taxpayers billions of dollars. As of September 30, 2018, the federal government had paid $7.4 billion in damages as a result of lawsuits and settlements, and its current estimate of remaining federal liabilities is over $28 billion. Every extra day of Congressional inaction costs American taxpayers another $2.2 million! Finally, U.S. commercial SNF inventories now exceed 80,000 metric tons at 98 operating reactors, 15 shutdown sites, and the Morris independent spent fuel storage site in Illinois, and all must be removed and permanently disposed per the federal government's obligation.

In closing, our organizations respectfully request that your response to our joint call for action include funding the noted priorities in FY 2020 and ensuring sustainable funding of an integrated nuclear waste management program. Thank you for your consideration, and please do not hesitate to contact Katrina McMurrian, NWSC Executive Director, at *katrina@theNWSC.org* or 615.905.1375, for more information.

Sincerely,

John E. Kelly
President
American Nuclear Society (ANS)

Wayne Norton
Chair, Decommissioning Plant Coalition (DPC) Steering Committee, and President & CEO, Yankee Atomic Electric Company

Nick Wagner
President, National Association of Regulatory Utility Commissioners (NARUC), and Board Member, Iowa Utilities Board

Maria G. Korsnick
President & CEO
Nuclear Energy Institute (NEI)

David Blee
President & CEO
U.S. Nuclear Industry Council (NIC)

Sarah Hofmann
Chair, Nuclear Waste Strategy Coalition (NWSC), and Commissioner, Vermont Public Utility Commission

cc: The Honorable Rick Perry, Secretary, U.S. Department of Energy
The Honorable Mick Mulvaney, Director, Office of Management and Budget

NRC NEWS
Office of Public Affairs, Headquarters
Washington, DC. 20555-0001
www.nrc.gov ■ opa.resource@nrc.gov

No: 19-021 May 7, 2019
CONTACT: David McIntyre, 301-415-8205

NRC Atomic Safety and Licensing Board Denies Evidentiary Hearing on Holtec International's Proposed Spent Fuel Storage Facility in New Mexico

A Nuclear Regulatory Commission Atomic Safety and Licensing Board has denied requests by several petitioners to hold an evidentiary hearing challenging Holtec International's license application to construct and operate a consolidated interim storage facility for spent nuclear fuel in southeastern New Mexico.

The three-judge board held oral arguments in Albuquerque in January on the standing of the various petitioners and the admissibility of their proposed contentions under NRC regulations. While the judges agreed that some of the six petitioners met the qualifications for standing, they concluded the nearly 50 contentions raised were not admissible for an evidentiary hearing. The judges held that the contentions either were not relevant to the application or did not establish a genuine dispute with aspects of the application.

The NRC's hearing process allows interested parties who might be affected by a proposed licensing action to challenge the application on technical (safety) or environmental grounds. Most hearings are conducted by licensing boards appointed from the Atomic Safety and Licensing Board Panel, a group of administrative judges independent of the NRC staff. Board rulings may be appealed to the Commission.

Holtec, a vendor of dry cask storage systems, has proposed to store spent nuclear fuel from the nation's commercial nuclear power plants at a facility in Lea County, in the southeastern corner of New Mexico. Hearing petitions were filed by Beyond Nuclear, the Sierra Club and the Fasken Land and Minerals and Permian Basin Land and Royalty Owners, which were granted standing. Two other petitioners – a coalition of several different organizations and NAC International, a rival dry storage cask vendor – were denied standing. The standing of a sixth petitioner, the Alliance for Environmental Strategies, was not decided.

The board's ruling, issued May 7, is available on the NRC's Electronic Hearing Docket on the agency website, under Holtec International 72-1051.

NRC NEWS
Office of Public Affairs, Region IV
Arlington, TX. 76011-4511
www.nrc.gov

No: IV-19-012 May 21, 2019
Contact: Victor Dricks, 817-200-1128

NRC Has Determined Fuel Loading Can Be Safely Resumed at San Onofre Nuclear Generating Station

The Nuclear Regulatory Commission has determined that fuel loading can be safely resumed at the San Onofre Nuclear Generating Station. The San Clemente, Calif., plant is owned by Southern California Edison and permanently shut down in 2013.

Fuel loading operations were suspended following an Aug. 3, 2018, incident involving a loaded spent fuel storage canister that was misaligned and became stuck on a flange while being lowered into a storage vault. Information about the incident and the NRC's response is available on the NRC website.

The NRC made its determination following extensive review of technical data submitted by Edison regarding the possible effects of scratching on spent fuel canisters during fuel loading operations.

The NRC will hold a virtual public meeting/webinar from 2-3 p.m. Central Time (12-1 p.m. Pacific Time) on June 3. Members of the public will have an opportunity to submit written comments and questions via the webinar user interface following a presentation by NRC officials. NRC staff will provide participation guidance during the webinar.

Interested members of the public should register for the webinar on the NRC website, at which time a confirmation e-mail will be sent with details for joining the webinar via computer or mobile device. There is an option to listen via a phone bridge; however participants must first register for the webinar to obtain the phone bridge number.