

Generic safety issue number 191 (GSI-191) - status and research activities

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Abstract: The present article reviews Generic Safety Issue Number 191, “Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance.” A short history of the events leading up to its issuance is discussed so the importance of the issue can be stressed. Research conducted for the U.S. Nuclear Regulatory Commission (NRC) on the issue is presented. Based on the currently available data from the NRC’s research, it is recommended that additional tests be run, so that a risk-informed solution to this complicated problem may be found in a timely manner.

Keywords: generic safety issue 191; materials; corrosion; LOCA; containment sump

1 Introduction

One of the first and foremost responsibilities of a nuclear power plant is to protect the public from radiation exposure. As such, reactor safety is one of the most important issues for nuclear engineers. In September 1996, the U.S. Nuclear Regulatory Commission (NRC) issued Generic Safety Issue Number 191 (GSI-191), which is titled “Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance.”

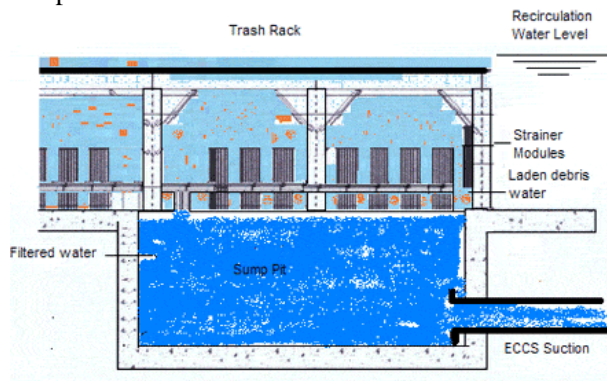


Fig. 1 PWR Containment Sump^[1].

This issue focuses on a very specific accident scenario and the potential effects of this scenario on the performance of safety related systems in pressurized water reactors (PWRs). The scenario starts with a loss-of-coolant accident (LOCA) in which primary water leaks into the containment, and broken piping,

insulation, and other debris could fall. The water and debris then gathers in the containment sump, where it is pumped back into the primary system via the emergency core cooling system (ECCS) to assist in cooling the reactor as it is shut down after LOCA. To protect the sump pump from any debris that may have been knocked loose, a screen filter is used. A diagram of a typical PWR’s containment sump is shown in Fig. 1^[1].

The hot, borated water ejected from the primary system increases the temperature and pressure within the containment, and contacts the various materials present in the containment building. This water is slightly acidic, and so is controlled with several types of pH control systems to make it more basic. Despite the pH control, the harsh environment produced by these effects will corrode the materials in the containment as well as the containment itself, which include aluminum, zinc, carbon steel, copper, fiberglass insulation, and concrete among others. The resulting corrosion products, along with any debris generated by the LOCA, could block and clog the strainer, leading to a loss of net positive suction head (NPSH) and lower the flow rate to the core that could, if left unchecked, lead to a core meltdown, and thus be a serious hazard to the public should any radiation be released.

The present article reviews this important reactor safety issue. First, a history of the accidents leading up to the issuance of GSI-191 will be presented. Next, several in depth studies will be reviewed. Finally, a summary of the important results will be presented, followed by recommendations for future research on the subject.

2 A history of GSI-191

The NRC became aware that a potential loss of NPSH could occur following a LOCA in 1985, and sent a generic letter to all licensees of all operating reactors^[2]. In this letter, the general problem is outlined, but no new requirements were imposed on reactor owners.

In 1992, a LOCA occurred at the Barsebäck Nuclear Power Plant's second unit, a boiling water reactor (BWR), in Sweden. During a routine startup following a refueling outage, a leaking pilot valve caused a rupture disc at the outlet of a safety release valve (SRV) to open^[3]. It was later determined that the pilot valve had been incorrectly assembled. Immediately following the SRV opened, high-pressure steam was vented into the drywell. This steam jet was forceful enough to knock loose approximately 200 kg of insulation material, half of which was found to have been carried to the condensation pool by water and steam flow. One hour after the LOCA, a high pressure drop alarm sounded for the containment vessel spraying system (CVSS) pumps. One of these pumps cavitated due to a lack of NPSH one hour after this alarm. It is important to note that the clogging of emergency system pumps had been taken into account by the reactor operators, but it was believed that there was no danger of it occurring until at least ten hours into an event. To make sure that the core could continue to be cooled by CVSS pumps, the system was backflushed so the clogs could be cleared. Analysis of the event found that the small strainer area was the main cause for the clogging taking place so quickly. After this event, the Swedish Nuclear Power Incorporate revoked operating permission for the five oldest BWRs, until the strainer issue could be resolved.

One year later, two separate events occurred at Perry Nuclear Power Plant, also a BWR, outside of Cleveland, Ohio. In an incident on 22 January, a blown fuse in the control circuitry caused a low water level alarm, which spurred an unplanned shutdown of the plant^[4]. During the event, the emergency core cooling system (ECCS) was plugged with suppression pool particulate matter^[5]. A few months later, the strainers were blocked again, but this time by fiberglass insulation that had fallen into the suppression pool^[6]. It was also noticed that a thick sludge had formed from where corrosion products had been strained by the fibers. In both events, the strainer was badly damaged by the pressure drops created by the blockages.

All three of these events caught the attention of the NRC, who issued a bulletin to BWR licensees to take measures to address the potential for ECCS clogging due to debris^[7]. It was found that the problem could be avoided by removing any fibrous material not rated to withstand LOCA conditions and by enlarging the strainer surface area^[8].

Despite the ease of the solution for BWR licensees, the same was not true for PWR licensees. Although no accidents of this type were recorded in PWRs, the NRC saw that the same issue could occur in the sump of PWR containments. Due to the differences in design between a BWR suppression pool and a PWR sump, the problem was not an easy fix for PWRs. As such, the NRC opened GSI-191 in 1996, so that a risk-informed solution to the problem could be ascertained.

After several years of further research on this subject matter, the NRC issued Generic Letter 2004-02^[9]. In this letter, the NRC required PWR plants to evaluate the recirculation functions of their ECCS and containment spray systems, and outlined some of the results that led them to force the plants to undertake the evaluation. It is this requirement that has brought GSI-191 related research to the forefront for PWR owners.

3 Relevant studies

The first major details of relevant data related to GSI-191 regard the description of the containment environment following a LOCA. Water from the primary systems of the reactor is ejected at temperatures of a maximum of 315 °C and at pressures of 15170 kPa (2200 psi). In the transient process, the maximum sump temperatures can approach 130 °C and the maximum gauge pressures can reach 248.2 kPa (36 psi). Once the containment environment has reached equilibrium, temperatures of up to 55 °C and atmospheric pressures are generally observed^[10]. As for the water itself, PWRs utilize boric acid in the primary water loop for reactivity control, so the initial water is slightly acidic. The pH of the primary water is balanced by the rate of boric acid and lithium hydroxide injection through the chemical shim system. In the event of a LOCA, the chemical shim system will not be functional, so the acidity needs to be controlled in a different manner within the containment. This containment pH control chemistry is plant specific, and may contain either lithium hydroxide or sodium hydroxide as the major pH controller, which is introduced via spray systems, or have buffers of either hydrated trisodium phosphate (TSP), or hydrated sodium tetraborate that exist as dissolvable powders in the sump^[11]. This harsh environment is difficult to incorporate into experiments.

In the event of an LOCA) within the containment of PWR, piping thermal insulation and other materials in the vicinity of the break will be dislodged or corroded by break jet impingement. Some of this debris and corrosion products can be accumulated on the suction sump screens of the emergency core cooling system (ECCS) pumps. The GSI-191 study addresses the issue of debris and corrosion products accumulation on the PWR sump screen. Various thermodynamic simulations were run to support GSI-191 research^{[12][13][14]}. In general, the thermodynamic simulations were able to perform several calculations:

- Aqueous speciation and saturation calculations
- Precipitation reactions
- Calculations at standard and elevated temperatures
- Aqueous processes at ionic strengths up to 0.5
- Maintain fixed conditions if deemed necessary

These types of calculations can be performed by several different thermodynamic codes, so runs in each code were made and compared. Although there were differences in the code outputs, the order of magnitude for each output was usually similar. From these various simulations, the expected species expected to precipitate out of solution were determined. All the results by thermodynamic calculations indicate that the major contributors to potential sump clogging are due to silicates, phosphates and hydroxides.

In 2005, the USNRC and the Industry (through EPRI) developed a joint 30-day Integrated Chemical Effects Test (ICET) program. Total five integral chemical effects tests were run. These tests were able to recreate the containment sump environment at the post-LOCA steady state conditions of 55 °C and atmospheric pressure. Corrosion was facilitated, and corrosion product concentrations in the solution were measured as a function of testing time. Various precipitates were identified. In fact, a gel-like substance was found at the bottom of the reaction tank following tests that incorporated TSP as the main pH buffer. It was determined that this gel was mainly composed of $\text{Ca}_3(\text{PO}_4)_2$ ^[11]. In general, the majority of the precipitates that were found were from the fiberglass insulation itself, or from the aluminum. It was also determined that the silicon and calcium from the fiberglass inhibited corrosion in the aluminum, thus limiting precipitate formation in the high-pH tests^[11]. In addition to characterizing the chemistry of the solution following 30 days of tests, the particle size distributions in solution were also analyzed. The distributions were in the range of 1 to 100 µm, and the mean particle size varied with each test.

In addition to these large-scale loops, tests were also conducted in small benchtop loops to measure the potential head loss across the sump strainer. Pacific Northwest National Lab (PNNL) operated multiple tests to relate any chemical effects to the measured pressure drop^[15]. At most, the pressure drop was measured to be 15.16 kPa (2.2 psi), with minimum drops of 2.068 kPa (0.3 psi). It was found that these pressure drops were extremely dependent on the material preparation, what order the debris material is placed on the screen, and the amount of material placed on the screen. The approach velocity also played a large role in the variability of the pressure drop. Although a pressure drop was measured, the drop (2.2 psi) is very small, and may not lead to an ECCS failure like that seen at the Barsebäck plant.

Based on the results, or rather lack thereof, from the ICET and PNNL experiments, some major factors could have been overlooked in the past NRC funded reports. Since most of the tests have been isothermal, it is postulated that the temperature transient seen during a LOCA could be such a missing factor. The widely accepted method used to address the chemical-effects for GSI-191 is to follow the approach recommended in a Westinghouse report titled “Evaluation of Post-Accident Chemical Effects in

Containment Sump Fluids to Support GSI-191” (WCAP-16530-NP), which has been approved by the NRC^[16]. Their approach had several steps. The first step was to run an integrated chemical effects test to obtain fundamental data on the corrosion processes in the post-LOCA environment. The next step was to identify those plant-specific materials that would influence the production of precipitates the most. The information from both of these steps would influence the third step, which was to run benchtop tests to develop information on which chemical products, and how much of these products, to use in the final step. In this final step, the sump screen performance would be evaluated. All of these tests were run at several pH levels (ranging from 4 to 12) and temperatures (ranging from 21 to 132 °C [70 to 270 °F]) to generate the material release and pressure drop correlations given in equations 1 through 4. Equation 1 describes the release rate of aluminum and galvanized steel.

$$\log(RR) = A + B(pHa) + C(1000/T) + D(pHa)^2 + E(pHa)(T)/1000 \quad (1)$$

In this equation, RR is the release rate in mg/m²/min, pHa is the temperature-corrected pH, T is the temperature in K, and all other symbols are constants to fit the data. These constants are supplied in Table 1.

Table 1 Constants in WCAP Release Rate Equations

Material	A	B	C	D	E
Aluminum	14.69039	0.00000	-4.64537	0.044554	-1.20131
Galvanized Steel	-15.10693334	-3.670953896	0.103589245	7.303961651	5.485050709

Equation 2 describes the release rate of silicates.

$$RR = kA(1 - C/K) \quad (2)$$

Here, k is a constant dependent on pH and temperature, A is the exposed surface area, C is the concentration of the released species, and K is the saturation limit of the released species. K and k are derived from equations 3 and 4, respectively.

$$\log(K) = a + b(pH) + c(1000/T) \quad (3)$$

$$\log(k) = d + e(pH) + f(1000/T) \quad (4)$$

a, b, c, d, e, and f are all constants that depend on the released species and the original silicate that it came from. These coefficients are given in Table 2.

Table 2 Coefficients to Determine Saturation and Rate Constants^[16]

Silicate	Released Species	Saturation Constant, K			Rate Constant, k		
		a	b	c	d	e	f
Calcium Silicate	Ca	-2.4063	-0.17595	1.967023	-2.35331	-0.15044	1.820687
Calcium Silicate	Si	0.12735	0.03197	0.71658	7.55470	-0.04084	-2.02198
Concrete	Ca	-0.15969	-0.04542	0.95477	5.31705	-0.07459	-1.10803
Concrete	Si	1.05597	0.01483	0.11862	3.50061	-0.01713	-0.74261
Concrete	Al	2.35338	0.06829	-0.70953	9.23778	0.05404	-3.34577
E-Glass	Ca	1.82949	0.06821	-0.47088	3.67611	0.02616	-0.96191
E-Glass	Si	5.20122	0.10404	-1.50553	7.46511	0.16247	-2.55813
E-Glass	Al	3.72351	0.14041	-1.69396	10.35371	0.17064	-4.17804
Min-K	Si	1.17043	0.10511	-0.07315	7.41106	0.17893	-1.93332
Aluminum Silicate	Si	7.51336	0.18619	-2.89181	7.17588	0.11502	-2.42532
Aluminum Silicate	Al	5.52900	0.24010	-2.51326	8.48062	0.20749	-3.32039
Mineral Wool	Ca	2.30159	0.12022	-0.82549	1.98549	0.09009	-0.52443
Mineral Wool	Si	5.95046	0.06796	-1.43151	6.07665	0.16569	-2.17413
Mineral Wool	Al	8.96613	0.10871	-2.37200	6.62900	0.13222	-2.57256
Interam	Si	13.60515	0.18354	-3.81145	15.69692	0.34838	-6.05941

However, the calculated amount of corrosion products and material deposition are not consistent with the ICET tests.

In addition to integral tests, several tests were run that tested single materials. These tests generated data for corrosion rates in a simulated post-LOCA containment environment. Jain, in addition to running thermodynamic simulations, also ran single material tests to verify corrosion rates^[12]. In his tests, he used a glass cell and autoclave systems to heat a borated water solution with a pH of 10 to different temperatures. The corrosion rates were obtained using polarization resistance measurements. Since concrete and fiber insulation do not corrode in the electrochemical sense, the leaching rates for their constituent materials were measured instead.

These corrosion and leaching rates of each material save for carbon steel agreed well with the thermodynamic data, and with data given by others, such as Griess and Bacarella^[17], Piippo, *et al.*^[18], Niyogi, *et al.*^[19], Hall^[20], Jantzen^[21], and Pan, *et al.*^[22]. Carbon steel's corrosion rate differed because Hall^[20] found that carbon steel was attacked much more aggressively by the pH control spray than by the

borated water. Since Jain's experiments were run in borated water, these rates are expected to be different.

Lahti *et al.*^[23] reviewed specifics on the corrosion processes and possible corrosion products of each of the materials present in the containment. Based on this review, the potential corrosion products for each important material are as follows:

- Zinc ions form the tetrahydroxozincate ion in high pH solutions. This ion may interact with any alkali metals in solution to create insoluble zinc oxides.
- Aluminum corrosion would only produce insoluble oxides or oxyhydroxides at temperatures greater than 60 °C, or at the beginning of a transient caused by a LOCA.
- Carbon steel dissolution would produce an unstable passive layer of insoluble iron hydroxide that could corrode to create an autocatalytic reaction. In addition, pitting corrosion may take place due to the presence of chloride ions. Both reactions could lead to a high rate of ferrous oxide production.
- Like carbon steel, the passive film of copper oxides could corrode to produce insoluble copper hydroxides. Pitting corrosion is also

likely in the presence of chloride ions, which leads to insoluble copper chlorides.

- Calcium ions can be leached from concrete, which could form calcium hydroxides. Chunks of concrete that may have been dislodged following a LOCA pose a bigger threat.
- In addition to clogging the strainers directly after becoming dislodged, silicon dioxide may also be leached from fiber insulation.

4 Conclusions and future research

Based on the history of and the available research relating to GSI-191, it is evident that this is a major issue in reactor safety. With the guidance of the NRC, nearly 20 years of research has been conducted, and a plethora of useful data regarding high temperature corrosion has been generated. Although this data is very helpful in quantifying the amount of corrosion products produced, and how much the resulting loss of NPSH is, no test has been able to accurately reproduce the pressure drops seen in the events at Barsebäck and Perry. To truly resolve GSI-191 in a risk-informed manner, new research must be undertaken. It is recommended to build upon the tests run by UNM^[10] and by Jain^[11]. Both tests were run at constant temperature, and only the ICET loop, incorporated the flow conditions expected in the containment sump. Thus, new tests should mimic the temperature transient seen during a LOCA, or, if the tests will be isothermal, other phenomenon that effect corrosion such as flow velocity, should be tested.

References

- [1] “Function of the Containment Sump.” U.S. NRC. U.S. NRC, 9 May 2013.
<<http://www.nrc.gov/reactors/operating/ops-experience/pwr-sump-performance/function-containment-sump.html>>.
- [2] “Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage.” Nuclear Regulatory Commission Generic Letter, GL85022 (1985).
- [3] “Updated Knowledge Base for Long Term Core Cooling Reliability.” Organization for Economic Co-operation and Development Report, NEA/CSNI/R(2013)12/ADD1 (2013).
- [4] CLAYTON, Zack: “January Monthly Report.” Ohio EPA, Letter to Jim Mehl (1 February 1993).
- [5] “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors.” Nuclear Regulatory Commission Bulletin, BL 2003-01 (2003).
- [6] Federal Register, Vol. 60, No. 146, 1995: 39021-39026.
- [7] “Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors.” Nuclear Regulatory Commission Bulletin, NRCB 96-03 (1996).
- [8] ZIGLER, G., *et al.*: “Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris”. NUREG/CR-6224 (1995).
- [9] “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors.” Nuclear Regulatory Commission Generic Letter, GL 2004-02 (2004).
- [10] RAO, D. V., *et al.*: “GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Pump Performance”. NUREG/CR-6762, Vol. 1. (2002).
- [11] DALLMAN, J., *et al.*: “Integrated Chemical Effects Test Project: Consolidated Data Report”. NUREG/CR-6914, Vol. 1 (2006).
- [12] JAIN, V., *et al.*: “Corrosion Rate Measurements and Chemical Speciation of Corrosion Products using Thermodynamic Modeling of Debris Components to Support GSI-191”. NUREG/CR-6873 (2005).
- [13] JAIN, V., *et al.*: “Chemical Speciation, Using Thermodynamic Modeling, During a Representative Loss-of-coolant Accident Event”. Report to the Nuclear Regulatory Commission, CWNRA 2004-07. (2004).
- [14] MCMURRAY, J., *et al.*: “GSI-191 PWR Sump Screen Blockage Chemical Effects Tests: Thermodynamic Simulations”. NUREG/CR-6912 (2006).
- [15] ENDERLIN, C. W., *et al.*: “Experimental Measurements of Pressure Drop Across Sump Screen Debris Beds in Support of Generic Safety Issue 191”. NUREG/CR-6917 (2006).
- [16] “Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191.” Westinghouse Owners Group Report, WCAP-16530-NP (2006).
- [17] GRIESS, J. C. and BACARELLA A. L.: “Design of Reactor Containment Spray Systems – Part III: The Corrosion of Materials in Spray Solutions”. Report to

- Oak Ridge National Laboratory, *ORNL-TM-2412, Part III*. (1969).
- [18] PIIPO, J., *et al.*: “Corrosion Behavior of Zinc and Aluminum in Simulated Nuclear Accident Environments”. Report to the Finnish Center for Radiation and Nuclear Safety, *STUK-YTO-TR 123*. (1997).
 - [19] NIYOGI, K. K., *et al.*: “Corrosion of Aluminum and Zinc in Containment Following a LOCA and Potential for Precipitation of Corrosion Products in the Sump”. *Proceedings of the Second International Conference on the Impact of Hydrogen on Water Reactor Safety, Albuquerque, New Mexico, October 3-7, 1982*. Rockville, Maryland: Nuclear Regulatory Commission, 1982.
 - [20] HALL, J. F.: “Corrosion of Low Alloy Steel Fastener Materials Exposed to Borated Water”. *Proceedings of the Third International Symposium on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors*. G. J. Theuss and J. R. Weeks, eds. La Grange Park, Illinois: American Nuclear Society, 1988: 711-722.
 - [21] JANTZEN, C. M.: “Radioactive Waste – Portland Cement Systems: II, Leaching Characteristics”. *Journal of the American Ceramic Society* 67.10 (1984): 674-678.
 - [22] PAN, Y., *et al.*: “Degradation of High-Level Waste Glass Under Simulated Repository Conditions”. *Journal of Non-Crystalline Solids* 319, 2003: 74-88.
 - [23] LAHTI, Erik, *et al.*: “Material corrosion in a reactor containment sump following a loss-of-coolant accident.” *Progress in Nuclear Energy* 75, 2014: 1-9.