Westinghouse

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 940-8560
e-mail: greshaja@westinghouse.com

LTR-NRC-17-77

December 18, 2017

Subject: Response to Revised November 21, 2016, U. S. Nuclear Regulatory Commission Request for Additional Information for Westinghouse Electric Company Topical Report WCAP-17794-P, Revision 0, and WCAP-17794-NP, Revision 0, "10x10 SVEA Fuel Critical Power Experiments and New CPR Correlation: D5 for SVEA-96 OPTIMA3" (Proprietary/Non-Proprietary)

Enclosed are copies of the proprietary and non-proprietary versions of "Response to Revised November 21, 2016, U. S. Nuclear Regulatory Commission Request for Additional Information for Westinghouse Electric Company Topical Report WCAP-17794-P, Revision 0, and WCAP-17794-NP, Revision 0, '10x10 SVEA Fuel Critical Power Experiments and New CPR Correlation: D5 for SVEA-96 OPTIMA3.'"

This submittal contains proprietary information of Westinghouse Electric Company LLC ("Westinghouse"). In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Nuclear Regulatory Commission's ("Commission's") regulations, we are enclosing with this submittal an Application for Withholding Proprietary Information from Public Disclosure and an Affidavit. The Affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Westinghouse is providing the proprietary information in this submittal to the Commission, for its use. In the event the Commission determines that any other U.S. or foreign government officials, outside of the NRC, are properly and directly concerned with the information in this Request for Additional Information (RAI) response, such that they must be given the opportunity to inspect the confidential and proprietary information in it, then Westinghouse requests that the Commission notify Westinghouse of that determination and engage with Westinghouse to develop an appropriate protective agreement to govern such a review, consistent with 10 CFR 2.390(b)(6).

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference AW-17-4684 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

J A Gresham

James A. Gresham, Manager
Regulatory Compliance

Enclosures

cc: Ekaterina Lenning
Dennis Morey

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 940-8560
e-mail: greshaja@westinghouse.com

AW-17-4684

December 18, 2017

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Response to Revised November 21, 2016, U. S. Nuclear Regulatory Commission Request for Additional Information for Westinghouse Electric Company Topical Report WCAP-17794-P, Revision 0, and WCAP-17794-NP, Revision 0, "10x10 SVEA Fuel Critical Power Experiments and New CPR Correlation: D5 for SVEA-96 OPTIMA3" (Proprietary)

Reference: Letter from James A. Gresham to the Document Control Desk, LTR-NRC-17-77, dated December 18, 2017.

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit AW-17-4684 signed by the owner of the proprietary information, Westinghouse. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the proprietary aspects of this Application for Withholding or the accompanying Affidavit should reference AW-17-4684 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

James A. Gresham, Manager
Regulatory Compliance

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, James A. Gresham, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse") and declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

Executed on: 12/18/17

James A. Gresham, Manager
Regulatory Compliance

(1) I am Manager, Regulatory Compliance, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Response to Revised November 21, 2016, U. S. Nuclear Regulatory Commission Request for Additional Information for Westinghouse Electric Company Topical Report WCAP 17794-P, Revision 0, and WCAP-17794-NP, Revision 0, '10x10 SVEA Fuel Critical Power Experiments and New CPR Correlation: D5 for SVEA-96 OPTIMA3'" (Proprietary), dated December 2017, for submittal to the Commission, being transmitted by Westinghouse letter LTR-NRC-17-77. The proprietary information as submitted by Westinghouse is that associated with the NRC review of WCAP-17794-P/WCAP-17794-NP, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to obtain NRC approval for the application of the D5 critical power ratio (CPR) correlation for Westinghouse SVEA-96 OPTIMA3 boiling water reactor (BWR) fuel assemblies

as documented in WCAP-17794-P/WCAP-17794-NP, Revision 0, "10x10 SVEA Fuel Critical Power Experiments and New CPR Correlation: D5 for SVEA-96 OPTIMA3."

(b) Further, this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of assisting customers in utilizing the improved design parameters of the CR 99 Control Rods.

(ii) Westinghouse can sell support and defense of the licensing of the D5 CPR Correlation and the SVEA-96 OPTIMA3 fuel design for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of a document, furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

Response to Revised November 21, 2016, U. S. Nuclear Regulatory Commission Request for Additional Information for Westinghouse Electric Company Topical Report WCAP-17794-P, Revision 0, and WCAP-17794-NP, Revision 0, "10x10 SVEA Fuel Critical Power Experiments and New CPR Correlation: D5 for SVEA-96 OPTIMA3" (Non-Proprietary)

December 2017

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

REVISED NOVEMBER 21, 2016, U. S. NUCLEAR REGULATORY COMMISSION REQUEST FOR ADDITIONAL INFORMATION FOR WESTINGHOUSE ELECTRIC COMPANY TOPICAL REPORT WCAP-17794-P, REVISION 0, AND WCAP-17794-NP, REVISION 0, "10X10 SVEA FUEL CRITICAL POWER EXPERIMENTS AND NEW CPR CORRELATIONS: D5 FOR SVEA-96 OPTIMA3" (TAC NO. MF3368)

NRR RAI

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]^{a,c}

Response to NRR RAI

Westinghouse has performed extensive inspections, investigations, and evaluations of the dryout events at **[** **]**^{a,c} Westinghouse has determined that the dryout events at **[** **]**^{a,c} were the result of specific operating conditions at **[**

]^{a,c} No correlation with the margin to dryout (as determined by the minimum critical power ratio) or the specific fuel design was evident. Detailed discussions of the inspections, investigations, and evaluations performed are presented in the following response.

CONTENTS

EXECUTIVE SUMMARY

Indications of a leaking fuel assembly (a Westinghouse SVEA-96 Optima2 fuel assembly) were detected at the **[ ]**[a,c]) in early July 2014 (towards the end of cycle 30). Following the examination of the leaking fuel assembly, it was concluded that dryout was the failure mechanism.

During the refueling outage from August to September of 2015, inspections of several fuel assemblies were performed. More extensive inspections were performed during 2016. The results of the visual inspections showed clear indications of locally increased oxide thickness and crud patterns on single fuel rods, attributed to localized dryout (LDO). Other than the one rod in the original leaking fuel assembly, none of the other inspected fuel rods with indications of LDO had failed.

The inspection records of SVEA-96 Optima2 and **OPTIMA3™** fuel assemblies operated at other nuclear power plants were reviewed. Fourteen fuel assemblies loaded in high risk locations in their first cycle of operation at another European BWR/6 were visually inspected. Fuel assemblies at 15 non-BWR/6 plants were also inspected. None of the fuel assemblies inspected, other than at **[ ]**[a,c], exhibited indications of dryout.

An assessment of the dryout failure in **[ ]**[a,c], based on extensive dryout testing and experience from dryout failures occurring at another plant in the late 1980's, concluded that the dryout indications on the affected rods in **[ ]**[a,c] are the result of a very localized dryout covering only a limited azimuthal sector of the rod.

In order to understand the causes of the localized dryout indications, Westinghouse performed a comprehensive root cause investigation, including exhaustive reviews and analyses of dryout and flow tests, plant operating data, occurrence of dryout indications versus critical power ratios (CPR), CPR methodology, and the dryout indication characteristics.

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]a,c

1 INTRODUCTION

Indications of a leaking fuel assembly were detected in **[ ]^{a,c}** in early July 2014 (towards the end of cycle 30). The plant continued its operation until the scheduled shutdown in early August. A detailed inspection of the leaking fuel assembly **[ ]^{a,c}** was initiated in February 2015, together with a number of measures to determine the failure cause. Following the examination of the leaking fuel assembly, it was concluded that dryout was the failure mechanism. In an attempt to reduce the risk of dryout, cycle 32 was operated at a higher flow rate, which normally would increase the margin to dryout. Instead, when inspections were performed at the completion of cycle 32, it was determined that more dryout indications had occurred during that cycle than during the previous cycle.

The sections that follow provide the details of the inspections and evaluations that have been performed to define and describe the dryout issue at **[ ]^{a,c}** Section 2 presents a summary of the Root Cause Analysis, Section 3 describes the **[ ]^{a,c}** (BWR/6) core support plate configuration, experimental verification of the flow conditions in plants with core support plate cross beam structures, the operating conditions present at **[ ]^{a,c}**, and the effect of **[ ]^{a,c}** Section 4 presents the inspection results from **[ ]^{a,c}** and other plants. Section 5 provides the evaluation of the **[ ]^{a,c}** dryout event. Section 6 provides the conclusions of the evaluations of the inspection data and extensive analyses and testing programs. Finally, Section 7 provides actions to be taken by Westinghouse for the transition to Westinghouse fuel for a U.S. BWR with a core support plate cross beam structure (BWR/6 design), as well as recommendations for the plant operator.

In addition, information is provided as appendices in support of the discussions presented in the body of this RAI response (Sections 1 through 7). Appendix 1 provides dryout experience with past dryout testing and actual fuel dryout failures that occurred at the Oskarshamn 2 plant. Appendix 2 discusses the applicability of the FRIGG test data in conjunction with the D4.1.5 CPR correlation used with the SVEA-96 Optima2 fuel design. Appendix 3 presents the Westinghouse response to the postulated impact of contributing factors offered to the NRC by GNF. Appendix 4 is a description of the inherent stability of the SVEA (SVEA-96 Optima2 and OPTIMA3) fuel design. Appendix 5 presents an independent investigation into the flow distribution in the SVEA fuel design. Finally, Appendix 6 presents the results of gamma spectrometry inspections of **[ ]^{a,c}** fuel assemblies.

2 ROOT CAUSES AND FAILURE SCENARIO

Westinghouse comprehensive Root Cause investigation can be summarized in the form of four key findings that enabled the identification of a combination of two root causes for the cooling deficiencies affecting fuel rods in several SVEA-96 Optima2 fuel assemblies, over several cycles, at **[** **]**[a,c] These findings are summarized as follows:

- **[**

][a,c]

Root Causes

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][a,c]

Also as part of this investigation, three substantive findings enabled to identify the key aspects behind the phenomenon affecting several of the peripheral rods in multiple SVEA-96 Optima2 fuel assemblies. These findings are summarized as follows:

- **[**

][a,c]

- [

]^{a,c}

Failure scenario leading to Localized Dryout

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The Westinghouse exhaustive Root Cause Analysis, scrutinized and reviewed on several occasions by both Westinghouse internal experts and external (non-Westinghouse) subject matter experts, is summarized in the following sections of this document. It should be pointed out that, to this date, no serious challenges have been brought up by any expert on the feasibility of the root causes or the failure mechanism identified during this investigation.

3 FUEL ASSEMBLY FLOW CONDITIONS IN BWR/6 PLANTS

In order to understand the nature of this event, it is necessary to get familiarized with some specific components affecting the coolant flow conditions in a BWR/6 plant. In particular, it is necessary to explore the specific flow conditions below the core support plate in a BWR/6 plant and those in the recirculation loops of a BWR utilizing jet-pumps (or diffusors) and having a center jet pump riser as a result of the design of these plants.

3.1 FLOW CONDITIONS AT SIDE ENTRY ORIFICE

A specific design feature of BWR/6 plants is the presence of a cross beams structure below the Core Support Plate (CSP). This structure creates a mesh of square cells where four (2x2) Control Rod Guide Tubes (CRGT) get surrounded by 1-inch thick and 2-feet high support beams. In the central (non-peripheral) locations of the core, on top of each of these Control Rod Guide Tubes, Fuel Support Pieces (FSP) are placed, with place for four (2x2) Fuel Assemblies (FA). Aligned orifices in both the CRGTs and the FSPs enable a path for the coolant to the individual fuel assemblies on top of the FSP. This is illustrated in the Figure 3-1. Since the flow paths through the Side Entry Orifices (SEO) are different, due to the presence of the cross beams, they have been given different nominations, in order to enable the use of different pressure loss coefficients for their hydraulic characterization in the 3D nodal codes. With this nomenclature, for example, SEO type 3 corresponds to the corner orifice facing two cross beams. As will be shown in the following sections, fuel assemblies loaded in "SEO 3" locations were particularly affected in this event.

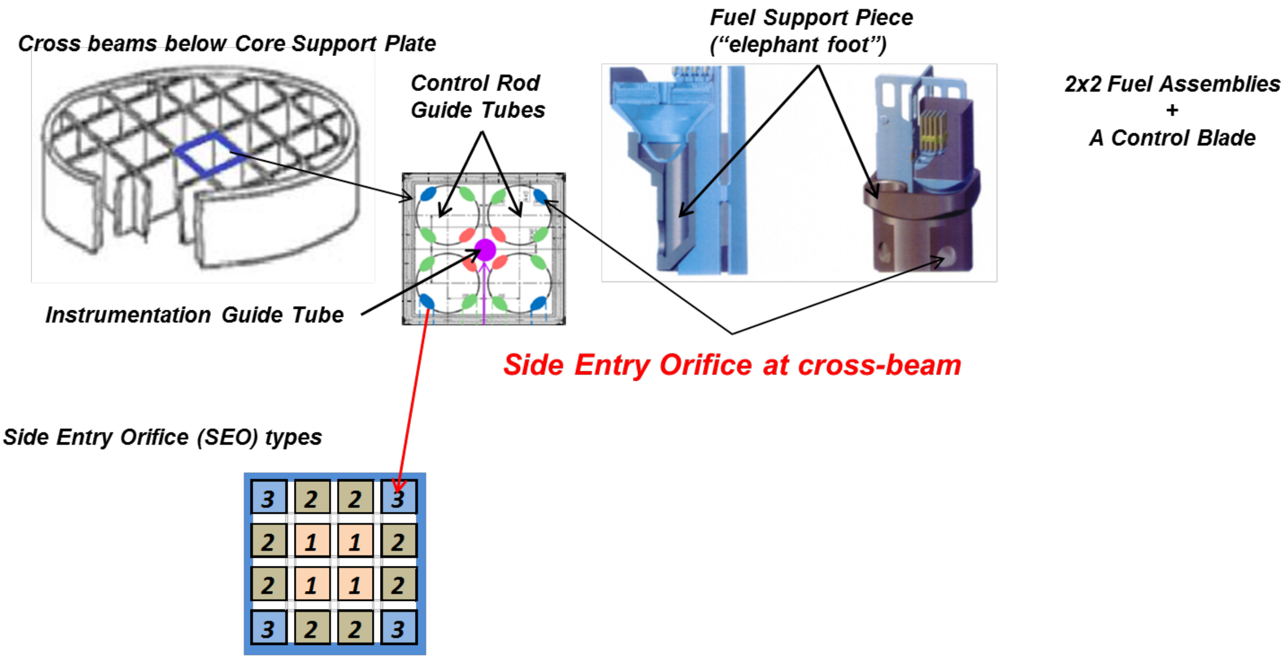

Figure 3-1 - BWR/6 design: Cross beams structure and Fuel Support Pieces

As shown in Figure 3-1, the cross beams structure defines a mesh where 16 (4x4) fuel assemblies are loaded. Four of them (at SEO 1) will not face the support beams, eight of them (at SEO 2) will face a single support beam, and the remaining four assemblies (at SEO 3) will

face the intersection of two support beams. The flow through SEO 3 will take the narrowest and most convoluted path of all.

On April 10th, 2002, a 10 CFR Part 21 Notification was filed at the U.S. NRC Operations Center Event Report regarding "Fuel support side entry orifice loss coefficient in Core Monitoring System Databank" Event # 39247. That notification was based on a Safety Information Communication (SC02-15) applicable to BWR/6 by GE Nuclear Energy and provided the following information regarding the fuel support side entry orifice:

1) *The core support structure for the identified plants affects the fuel support casting SEO loss coefficient. Bundles may be adjacent to zero, one, or two core support beams depending upon their location as illustrated in Figure 1. The SEO loss coefficient depends upon the number of adjacent core support beams. The core monitoring system in the affected plants uses an average SEO loss coefficient for all of the central bundles, and a separate average value for all of the peripheral bundles. This was previously evaluated for GE 8x8 fuel designs and found to be acceptable. Recent calculations have shown that the Critical Power Ratio (CPR) response for newer GE fuel designs are more sensitive to the reduced flow in bundles that are adjacent to two core support beams (which have the highest loss coefficient). Thus, the core monitoring system over predicts CPR for these bundles, and thereby, may under predict the margin to the Operating Limit Minimum CPR (OLMCPR). The CPR over prediction is approximately 0.01 in CPR …*

   …The SEO is the primary flow path for core flow to enter a fuel bundle. Due to the core inlet geometry in a BWR/6, the flow path to the SEO is more tortuous for a bundle that is adjacent to two core support beams. The bundles adjacent to two core support beams will have a loss coefficient that is about 20% higher than a bundle adjacent to one core support beam, and about 40% higher than a bundle that is not adjacent to any core support beams.

While SC 02-15 refers to the SEO loss coefficients in the core monitoring systems, the same (or very similar) codes and input data are normally used for the nuclear and thermo-hydraulic core design and in the determination of the plant operating limits.

3.2 INVESTIGATIONS ON THE STABILITY OF THE FUEL ASSEMBLY INLET FLOW

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]a,c

1) "Loss Coefficients Determination for Side Entry Orifices through CFD Simulations" - Jin Yan, Jens Andersen, Brian Golchert (GE-Hitachi Nuclear Energy) - Transactions of the American Nuclear Soc., 2007, vol. 97, pp. 413-414.

2) "Experimental and Numerical Investigations of BWR Fuel Bundle Inlet Flow" - E. Hoashi, et.al. (TOSHIBA Corporation) – NURETH-13, 2009 - N13P1358.

3) "Role of complicated flow fields in lower plenum on coolant flow distribution to core of ABWR" – Shun Watanabe, Yutaka Abe, Akiko Kaneko, Fumitoshi Watanabe, Kazuki Hirao – ICONE20-POWER2012-54848

The following is an excerpt from the first reference mentioned above:

> *"… It was found that the flow rate and the pressure loss for four fuel bundles show noticeable differences depending on the locations of the orifices relative to the support beams. Complicating the analysis was the fact that there was significant unsteadiness in the flow field. This unsteadiness was indicated by a fluctuating mass flow rate through each bundle."*

Furthermore, of particular importance for the understanding of this phenomenon are the experimental investigations **[**

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]a,c

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]a,b,c

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]a,b,c

The inherent instability of the flow vortices at specific SEO locations was clearly demonstrated [ ]a,c Moreover, the combined evaluation [ ]a,c, highlights the significant impact of the flow field in the lower plenum on the flow conditions at the inlet to the fuel assemblies, something neglected in the original evaluation performed by the original equipment manufacturer (OEM).

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]a,c

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]a,b,c

3.3 CFD INVESTIGATION OF THE FLOW CONDITIONS AT THE SIDE ENTRY ORIFICE

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]a,b,c

3.4 CORE FLOW – BI-STABLE CONDITION IN RECIRCULATION LOOPS

The two recirculation loops at **[ ]^{a,c}** are based on recirculation pumps feeding 10 jet-pumps each, pair-wise fed through five "risers". The possibility of a bi-stable flow condition developing in a recirculation loop flow in jet-pump reactors is well-known by the Utilities operating such plants. [1] USNRC Information Notice No 86-110 "Anomalous behavior of recirculation loop flow in jet pump BWR plants", [1], briefly described this occurrence.

In a Service Information Letter SIL No. 467, [2], issued by the plant vendor (GE) in 1988, a bi-stable flow has been attributed to random occurrences of fluctuations in recirculation system drive flow during steady-state operation. The frequency of change and duration of each flow pattern appears to be sensitive to small changes in influencing parameters such as the recirculation flow rate, the flow control valve position and the pump speed. Consequently, the specific plant design and operating conditions are of crucial importance for the occurrence and magnitude of the bi-stable conditions.

In SIL No. 467, the OEM concluded that the cause of the observed changes in the recirculation flow is a bi-stable flow pattern at the jet pump header cross in the recirculation pump discharge piping that reduces the flow to the outer risers. No occurrence of bi-stable flow has been reported for BWRs that do not have a center jet pump riser. The redistribution of the drive flow to the jet pump risers is a characteristic of the bi-stable flow which distinguishes it from other events that can increase or decrease recirculation flow. As described in Section 3.5, the total core flow at **[ ]^{a,c}** was not significantly affected by the bi-stable condition.

The OEM recommends that the plant owners, upon experiencing this anomalous operating condition, perform a plant-specific evaluation of the potential consequences for fuel assemblies and other components.

The main conclusions of the SIL can be summarized as:

- Bi-stable flow has been observed under a variety of conditions in BWR/3s through BWR/6s.

- Some BWRs do not have a center jet pump riser. Owners of BWRs, which do not have the header cross design, have reported no occurrences of bi-stable flow to GE.

- The observed frequencies ranged from one to 200 cycles per hour.

- The frequency of change and the duration of each flow pattern are random, and appear to be sensitive to small changes in influencing parameters, such as recirculation pump flow rate, flow control valve positions, and pump speed.

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]^{a,c}

[1] In a dynamical system, bi-stability means the system has two stable equilibrium states. In terms of potential energy, a bi-stable system has two local minima of potential energy separated by a peak (local maximum). The system can rest in either of the two states and, if perturbed, the system can transition between the two states.

3.5 FLOW CONDITIONS IN [ ]a,c RECIRCULATION LOOPS

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[2] [

]a,c

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]a,b,c

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]a,b,c

These observations are consistent with the conclusion of an increased risk for the coolant disturbances occurring at higher total core flow levels.

Note: Recent inspections of fuel assemblies first loaded in Cycle 33 did not show any indication of increased oxidation. Cycle 33 operated with an explicit flow restriction of 95%

of rated total core flow, **[**

]^{a,c}

3.6 CONCLUSIONS OF INVESTIGATION OF FLOW CONDITIONS AT **[** **]^{a,c}**

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]^{a,c}

4 INSPECTION RESULTS AND EVALUATION OF OBSERVATIONS

4.1 **[ ]**^{a,c} INSPECTIONS

During the refueling outage in August – September 2015 inspections were performed; a total of 19 SVEA-96 Optima2 fuel assemblies (FAs) were inspected in addition to **[ ]**^{a,c} More extensive inspections were performed during 2016 including 186 fuel assemblies. Two of those fuel assemblies had also been inspected in 2015.

The results of the visual inspections in 2016 showed the same type of indications as in earlier inspections: clear indications of locally increased oxide thickness and crud patterns on single fuel rods below spacer grids 7 and 8, attributed to localized dryout (LDO) in several FAs. The dryout indications occur only on the rods next to the 1/3 part-length corner rods. In none of the 2016 inspections was the integrity of the affected fuel rods compromised.

The number of affected rods in each fuel assembly ranges from one to six. In some cases fuel rods in all four sub-assemblies were affected. **[ ]**^{a,c}

The observations occur more frequently below spacer grid 7. Most rods with indications below spacer 8 also have indications below spacer 7. The length of the marks ranges from a few millimeters up to approximately 250 mm. The increased oxidation mark on the leaking **[ ]**^{a,c} had an approximate length of 260 mm.

[ ]^{a,c}

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]a,b,c

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]a,b,c

[3] [ ]a,c

4.1.1 Affected Fuel Assemblies Operating Conditions Characterization

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]a,b,c

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]a,c

[4] [ ]a,c

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]^{a,b,c}

4.2 STATISTICAL ANALYSIS OF [ ]^{a,c} INSPECTIONS – CORE POSITIONS

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]a,b,c

4.3 INSPECTIONS OF SVEA-96 OPTIMA2 AND 3 AT OTHER PLANTS

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]a,c

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]a,b,c

5 EVALUATION OF THE DRYOUT EVENTS

5.1 THE DRYOUT FAILURE IN [ ][a,c]

An assessment of the dryout failure in [ ][a,c] was made based on the information summarized in Appendix 1 of this document regarding the dryout tests in the Halden test reactor (IFA-157 and IFA-613), the PWR cladding corrosion test (IFA-708 in the Halden test reactor) where a dryout event unintentionally occurred, earlier experience from dryout failures in Oskarshamn 2 in 1988 and studies of effects of temperature excursions on cladding properties. The conclusions of the assessment are:

1) Severe damage to a fuel rod during a short term transient requires very high cladding temperatures (~1000 °C) and would result in clad collapse (instead of ovality, as observed in the [ ][a,c]

2) Degradation of clad properties during a short term transient that could lead to fuel failure during the subsequent normal operation requires temperatures above the α to α+β phase transformation temperature (about 800 °C), and would result in collapse (not ovality).

3) Corrosion during dryout resulting from lower clad temperatures requires (at least) several days to penetrate the cladding.

4) An increased cladding ovality, in combination with an observed increased oxidation, indicates very local overheating as observed in IFA-708.

5) The most likely identified scenario that can explain the observations made from the affected rods in [ ][a,c] is the onset of a very localized dryout (covering only a limited azimuthal sector of the rod), maintained over longer periods and possibly interrupted by consecutive re-wettings, thus resulting in relatively modest cladding temperatures (well below 800 °C and likely below 700 °C).

5.2 STATISTICAL ANALYSIS OF [ ][a,c] INSPECTIONS – MINIMUM CPR

[

]

[a,c]

[

]^{a,b,c}

As can be seen, both distributions are quite similar without any indication of significantly increased frequency of oxidation at lower margins to dryout. This was confirmed by a more stringent statistical evaluation **[**

]^{a,c}

In conclusion, the lack of correlation between dryout margin and increased oxidation marks occurrence frequency is statistically significant and demonstrates that the film evaporation alone is not the cause of the **[ ]**^{a,c} dryout event.

5.3 SVEA-96 OPTIMA2 OPERATING EXPERIENCE

SVEA-96 Optima2 is a mature fuel product, with approximately 12,000 fuel assemblies delivered so far over more than 15 years to 19 BWR plants in both Europe and USA. The nuclear design of the affected fuel assemblies at **[ ]**^{a,c} does not deviate significantly from other fuel assemblies in operation. It has a typical enrichment and burnable absorber design, similar to the designs that have been used in previous unaffected cycles at **[ ]**^{a,c} and at other unaffected plants including another European BWR/6 plant. In all cases, the CPR margins were determined using the same approved calculation methods and methodologies.

The affected fuel assemblies at **[ ]**[a,c] had minimum CPR margins, and operating at assembly power and flow levels, in the same range as most of all the delivered SVEA-96 Optima2 assemblies. **[**

][a,b,c]

Based on this comprehensive operational experience as well as the consistent accuracy of our and [ ]a,c method predictions, it is not considered credible that a generic fuel design deficiency (or combination of several ones) could have led to a margin erosion extensive enough to result in dryout. In such case, frequent and/or more extended dryout events would have been observed earlier.

Note: Since the compilation of the data for Figure 5-2, additional operating experience has been obtained at several NPPs. However, since no dryout observations have been identified at any other plant, this additional data would only reinforce the conclusion. Furthermore, considering that oxidation marks were found in FAs with a minimum CPR [ ]a,c, the operating experience base would include almost all of the 12,000 fuel assemblies delivered to date.

5.4 CONCLUSIONS REGARDING THE [ ]a,c FAILURE MODE

[

]a,c

[

]a,c

Several independent researchers, among the highest qualified thermal-hydraulics experts, have been consulted by Westinghouse in order to get their view on the key components of the above-described conclusions. Based on their experience and knowledge, each of them has concluded that the basic mechanism described is both feasible and reasonable.

6 CONCLUSIONS

Westinghouse has conducted rigorous and detailed investigations of the circumstances surrounding the dryout events at [ ]^{a,c} An extensive root cause analysis has been completed.

Inspections have been performed, both of fuel assemblies at [ ]^{a,c} and assemblies at a number of other BWRs, including BWR/6 plants and other BWR designs, to determine the extent of condition of the dryout phenomenon at [ ]^{a,c}

The dryout indications have been studied with regard to the characteristics of classical dryout. Based on past dryout events and specific dryout testing, it is concluded that the dryout scenario experienced at [ ]^{a,c} is a localized dryout not characteristic of classical dryout.

[

]^{a,c}

Additional existing tests at Westinghouse flow loop facilities evaluated the possibility of flow instabilities between the sub-channels of the SVEA fuel design. This testing demonstrated the inherent stability of the SVEA design, eliminating the sub-channel design as a contributing cause to dryout.

An analysis of the fuel assemblies with dryout indications was performed to determine whether a correlation existed between minimum CPR and the frequency of dryout indications. No such correlation was found. Further investigation of Westinghouse BWR operating experience has also shown that many SVEA-96 Optima2 fuel assemblies have operated at lower MCPR than those at [ ]^{a,c} without any failures or dryout indication.

[

]^{a,c}

[

]a,c

[

]a,c

7 OPERATION WITH WESTINGHOUSE FUEL IN A U.S. BWR

[

]a,c

8 REFERENCES

[1] U. S. NRC Information Notice No. 86-110, "Anomalous Behavior of Recirculation Loop Flow in Jet Pump BWR Plants," December 31, 1986.

[2] GE Service Information Letter (SIL) 467, "Recirculation System Bi-stable Flow in Jet Pump BWRs," July 28, 1988.

APPENDIX 1: EXPERIENCE OF FUEL PERFORMANCE DURING AND AFTER DRYOUT

Experience of fuel behaviour during and after a dryout has been gained through dryout tests in the Halden test reactor in the late 1960's and early 1970's (IFA-157), as well as in the 1990's (IFA-613). In addition, dryout failures were experienced in Oskarshamn 2 in 1988 and most recently in an experiment with high steaming rate performed in the Halden test reactor (IFA-708).

HALDEN IFA-157 IN-REACTOR DRYOUT EXPERIMENT

In the late 60's and early 70's the irradiation experiment in Instrumented Fuel Assembly No. 157 (IFA-157) was carried out as a joint effort between ASEA Atom, AB Atomenergi and the OECD Halden Project. It was devoted to the study of long term behavior of Zircaloy cladding and UO2 fuel operating close to and under dryout conditions.

The IFA-157 test assembly consisted of 9 fuel rods in a 3x3 matrix. The cladding material was Zircaloy-2 in annealed and recrystallized condition.

The dryout experiments were performed at a constant power level by throttling of the inlet mass flow until one of the detectors indicated dryout. After dryout indication the mass flow was quickly increased by opening the valve. The reactor power was then changed and the same procedure was repeated several times. During the irradiation period (February 1968 until September 1971), dryout indications were achieved 81 times. The longest single dryout signal lasted for approximately 45 seconds. During the experiment dryout signals were recorded during a total of 1300 seconds.

Metallographic examinations of the claddings were made at peak power positions. These examinations showed in general a good and quite intact cladding with the exception that certain regions were heavily hydrided. A favourable feature of the heavily hydrided areas was that the inner part of the cladding was not affected. Thus the ductility reduction was probably low.

The metallographic examinations further showed that the grain size was not changed during irradiation and there were no sign of β-phase transformation during the transient. This indicates that the temperature in the cladding did not exceed 800°C. The data on diameter increase were nil or very small. Since the data contained crud and oxide measurements, it was doubtful whether any actual diametric clad strain had taken place. The fuel to cladding gap was unchanged.

It was concluded that the Halden data show that Zircaloy cladding withstands successive dryout sequences without severely changing cladding mechanical properties.

HALDEN IFA-613 IN-REACTOR DRYOUT EXPERIMENT

The objective of the dryout experiments in Instrumented Fuel Assembly No. 613 (IFA-613) was to provide information on the consequence induced on fuel by short term dryout events having characteristics similar to those anticipated to occur from pump trips in BWRs. In

particular, the post-dryout fuel performance was to be assessed both in terms of fuel behaviour during service after the test and in terms of fuel rod property changes as determined by post irradiation examinations.

A series of dryout experiments were carried out in IFA-613, followed by extensive PIE (Post-Irradiation Examination) of the rods, to study what effect in-pile transient heating, resulting from dryout, has on the microstructural and mechanical properties of irradiated cladding.

Two fresh rods and 6 rods pre-irradiated to 22-40 MWd/kg (Zr-2, Zr-2 with liner and Zr-4) were exposed individually to reduced or no-flow conditions in a heated light water loop within the Halden reactor. Each rod was instrumented with 2 or 3 surface thermocouples and a clad elongation detector and the estimated peak clad temperatures (PCTs) for 6 of the rods were in the range 950-1200 °C while PCTs of 750-850 °C occurred in the other two rods.

The 8 rods in the IFA-613 dryout test series experienced PCTs in the range 750-1150 °C and hold times in dryout conditions (not at maximum temperature) between 5 and 80 s. The 6 test segments in the first two loadings were irradiated for 24-30 days after the dryout experiments and none developed failure.

The conclusion of the dryout experiments in IFA-613 was that fuel rods can sustain dryout conditions for at least 80 s followed by consecutive normal steady-state operation for at least 30 days without developing any failure.

[

]^{a,c}

DRYOUT FUEL FAILURES IN OSKARSHAMN 2, 1988

In the beginning of 1988 four fuel rods in Oskarshamn 2 failed. The rods failed during their first cycle of operation. The cause could be identified as dryout brought about mainly by an excessive channel bow of neighboring assemblies [4].

Four fuel rods, one failed, one collapsed to pellet, one with burnable absorber (BA), and one reference rod were examined at Studsvik.

On the failed rod severe primary damage was seen in a region extending from just below the top spacer and 20-30 cm down. Profilometry showed cladding collapse onto the pellets and into pellet gaps over the region of visible primary damage. The region immediately below showed collapse and ridging. Metallography in the zone of visible primary damage showed that phase transformation had occurred. The oxide was very thick (> 200 µm). Hydrogen concentrations of 750-4400 ppm were detected. The highest concentration was, due to migration of hydrogen atoms to colder spots, seen on the side not affected by dryout. It was concluded that the total time at actual dryout was more than one day but less than one week.

The other rod, collapsed but not failed, had collapsed to pellet below the top spacer. Analysis of cladding from the collapsed zone showed that the cladding had remained in the α-phase. It was concluded that the cladding temperature had been above 550 °C, but below 800 °C.

The Oskarshamn 2 experience indicates that high cladding temperature must be sustained during several hours before severe cladding damage occurs.

[

]^{a,c}

[

]a,c

POST DRYOUT ZIRCALOY CLADDING PROPERTIES

The goal of the IFA-613 dryout test series was to further strengthen the basis for a fuel rod failure criterion based on clad temperature history. Rather than describing this type of criterion as a rod failure criterion it might, however, be visualized as a temperature-time space for the peak clad temperature within which the fuel rod can be regarded as unaffected by the temperature excursion. That is, no immediate fuel failure is expected if the clad temperature exceeds the limit set by the criterion. However, if the temperature excursion goes beyond the limit, the fuel rod might deviate from its design performance basis in its subsequent normal operation.

Prior to the IFA-613 dryout test series, the effects of high temperature (700 – 1000 °C) transients on corrosion properties, hydrogen uptake and mechanical properties of Zircaloy cladding material were studied by ABB Atom. Post dryout autoclave experiments indicate that the peak cladding temperature could reach values near the α to α+β Zircaloy phase transformation temperature (about 800 °C) during short temperature transients without severe changes of cladding post dryout oxidation properties.

Oxidation data from transients at 700 °C and 800 °C were used to generate an oxidation rate equation capable of predicting the extent of oxidation during a dryout in the temperature range below the α to α+β phase transformation temperature. Zircaloy yield strength was used to limit the maximum acceptable cladding temperature during a transient. For cladding temperatures below this limit, Zircaloy cladding creep limits the acceptable transient duration. The Gittus and Haste correlation for high temperature creep was used to evaluate clad deformation during a dryout event. Finally a temperature-time space for the peak clad temperature within which the fuel rod can be regarded as unaffected by the temperature excursion was proposed (Figure A1-1 below).

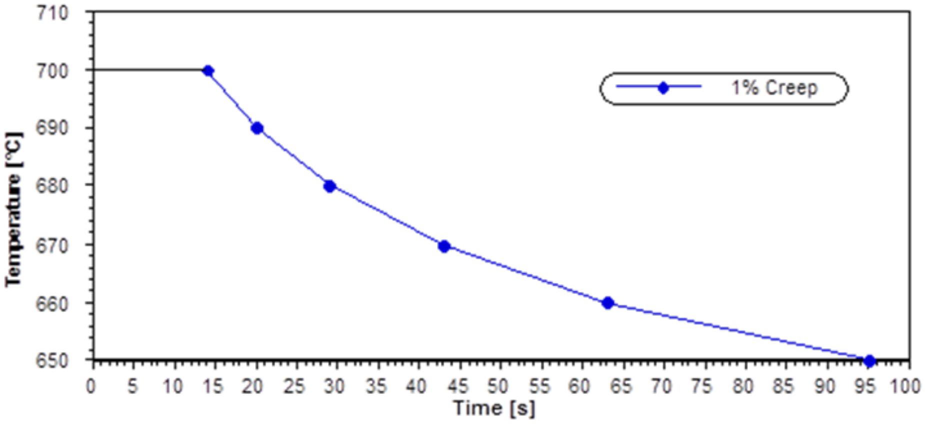


Figure A1-1 Combinations of times and temperatures that result in 1% circumferential creep strain, assuming a hoop stress of 47 MPa

It should be emphasized that the temperature-time curve in Figure A-1 describes the space within which the fuel rod can be regarded as unaffected by the temperature excursion. Exceeding the curve does not lead automatically to a prompt fuel rod failure.

APPENDIX 2: THE DRYOUT PERFORMANCE OF SVEA-96 OPTIMA2 AND OPTIMA3 - APPLICABILITY OF FRIGG DATA AND CPR CORRELATION

An extensive report regarding the accuracy and applicability of the D4.1.5 CPR correlation used for SVEA-96 Optima2 fuel was previously shared with NRC, **[ ]**^{a,c} This report is considered Westinghouse internal documentation not typically referenced in topical reports. However, it can be made available for audit by NRC personnel. This appendix provides a summary of that report.

VALIDATION OF THE TEST FACILITY

The FRIGG thermal-hydraulic test loop located in Västerås (Sweden) has been utilized since the 1960'ies for the qualification of all Westinghouse BWR fuel designs (both 8x8 and 10x10). The loop tests cover all requirements for BWR fuel heat transfer (dryout), pressure drop and thermal-hydraulic stability.

As part of a Technical Development Agreement between ASEA-ATOM[5] (AA) and General Electric (GE), a series of comparative full-scale dryout experiments were performed at the AA FRIGG and GE ATLAS loop facilities during the late 1970's. Comparisons between FRIGG and ATLAS data, and between correlations based on the corresponding data, were made. The data included measurements of critical power and pressure drop. The selected bundle was a 4×4 subset of the 7×7 lattice type with power peaking in the corners and cosine axial heat flux distribution. The measurement procedures were those specific to each test facility. Information regarding this validation was provided to the NRC in the answer to RAI-SNPB-02, submitted via LTR-NRC-16-53, August 8, 2016.

Direct comparison of the measured critical power levels showed highly consistent trends. **[ ]**^{a,c} A discrepancy of that order could well be explained by the combined measurement uncertainties in the two series of experiments and, hence, it was concluded that there was no evidence of any laboratory bias.

SUB-BUNDLE VERSUS FULL-BUNDLE TESTING

[

]^{a,c}

[5] ASEA-ATOM later become ABB Atom and is the same cognizant organization known as Westinghouse Electric Sweden today.

[

]a,b,c

[

]^{a,c}

APPROPRIATENESS OF DRYOUT CORRELATION

A careful re-assessment of the applicability and validity of the licensed CPR correlation for SVEA-96 Optima2 fuel has been performed considering the specific conditions at **[ ]^{a,c}.** The review focused on normal operating conditions at **[ ]^{a,c}**, as given by the plant technical specifications, and is based on the same conditions as used in the Reload Licensing Submittal analyses. More specifically, it is assumed that the flow to the inlet of each fuel assembly is stable over time. The evaluations were limited to such steady-state conditions.

The following evaluations were performed:

[

]^{a,c}

[

]^{a,c} It is concluded that the D4.1.5 CPR correlation is indeed appropriate for calculating CPR

margins of SVEA-96 Optima2 fuel in **[ ]**[a,c], when assuming conditions of steady-state operation (and AOOs) in accordance with the plant technical specifications.

REFERENCES

A2-1 **[**

][a,c]

APPENDIX 3: WESTINGHOUSE RESPONSE TO THE "GNF PERSPECTIVE ON OPTIMA-2 DRYOUT"

PREDICTED AND EXISTING MARGINS TO DRYOUT

It has been argued by GNF, that the **[ ]**[a,c] dryout observations are best explained by an ordering influence of multiple contributing factors and inherent weaknesses of the SVEA design. The estimated MCPR impact according to information provided by GNF to the NRC, (ADAMS Accession Number ML1721SA338) is purported to be described in the figure below, extracted from the mentioned reference:

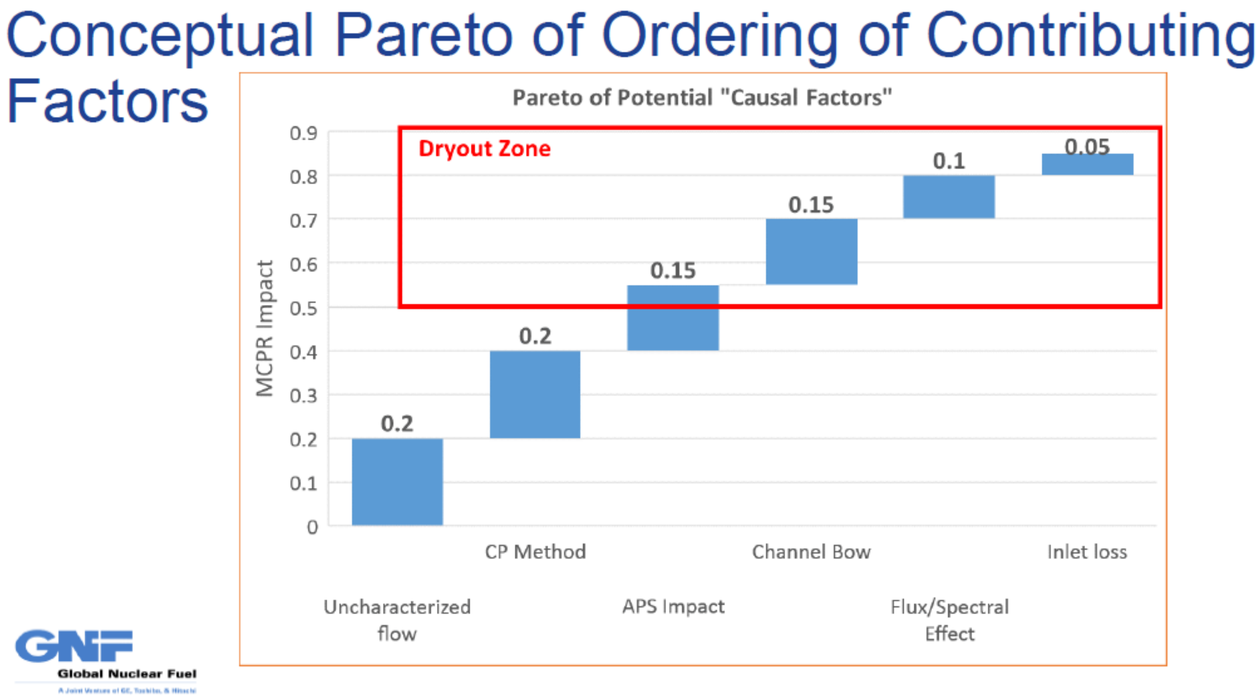


Figure A3-1: GNF's Conceptual Pareto of Ordering of Contributing Factors (extracted from ADAMS ML1721SA338)

All of the contributions presented in Figure A3-1 have been previously evaluated by Westinghouse using all of the information available about the specific conditions at **[ ]**[a,c] While Westinghouse agrees that these are all obvious factors to be considered based on past industry experiences, it has been demonstrated that they have no significant relevance to localized dryout observations, even when conservatively taken in combination. The relevance and potential magnitude of contribution of each factor are discussed below.

UNCHARACTERIZED FLOW – CPR ERROR ESTIMATED TO 0.20

This factor refers to the potential uneven distribution of flow between the SVEA sub-channels. This has been hypothesized **[ ]**[a,c] to occur either in steady-state operation, as a result of the asymmetric inlet conditions of the fuel support piece in a GE plant design, or

dynamically resulting from hydrodynamic instability between the SVEA sub-channels, causing the sub-channel flows to oscillate out of phase.

The dynamic instability effects are addressed in Appendix 4 which provides a detailed discussion of the stability properties of SVEA fuel as well as its inherent robustness against parallel sub-channel instability. It has been proven, through dedicated experiments at the FRIGG loop, independent experimental research and theoretical investigations as well as detailed computations, that the SVEA four sub-channel system with ventilated communication between the sub-channels is inherently stable. Parallel sub-channel instability, even under forced oscillations in the fuel assembly inlet flow, can therefore be excluded.

Further evidence that T/H instability cannot be the cause of the increased oxidation observed in **[ ]**^{a,c} is the fact that the plant has been operating at reduced power/flow conditions during a limited number of hours each cycle. The total time spent at off-rated power/flow conditions per cycle **[ ]**^{a,c} is significantly less than what is needed to explain the measured oxide thicknesses. **[ ]**^{a,c}

On the other hand, referring to the steady-state effects, **[ ]**^{a,c} that suggested that there was a flow mismatch at the inlet to each SVEA sub-bundle. **[ ]**^{a,c} In addition to having been carefully addressed at the introduction of the SVEA channel design, the concern was further addressed by a new dedicated test performed in the Westinghouse FRODE test loop as presented in Appendix 5.

The FRODE loop test demonstrated that

1) the flow mismatch at the sub-bundle inlet is much less than **[ ]**^{a,c}, and

2) the flow mismatch is completely evened out, through the communication slots between the sub-channels, over the first 0.5 meters, i.e. before entering the 2-phase flow regime.

Thus, the potential CPR impact is at most 0.01, and not 0.20 as claimed by GNF.

Further evidence that the flow mismatch has no relevance to the observed dryout indications comes from the fact that all four sub-bundles have been affected by dryout, even within the same fuel assembly (e.g. the failed fuel assembly **[ ]**^{a,c}). In fact, the dryout failure in **[ ]**^{a,c} was in the sub-bundle receiving the highest inlet flow according to both the CFD calculations and the FRODE measurement.

CRITICAL POWER (CP) METHOD (INCLUDES TEST CONDITIONS IN THE FRIGG LOOP) – CPR ERROR 0.20

This was the obvious presumed failure cause and Westinghouse initially focused on this area. After careful investigations, Westinghouse did not find anything more than what is already

included in the uncertainties considered for the determination of the plant operating limit MCPR. [ ]^{a,c}

An extensive report regarding the accuracy and applicability of the Critical Power (CP) methods [ ]^{a,c} is summarized in Appendix 2. Generally, fuel vendors use very similar CP methods and dryout testing conditions.

The FRIGG loop was validated against the GE ATLAS loop by performing dryout tests on the same fuel design at both locations. This was previously discussed in the response to RAI-SNPB-02 submitted via LTR-NRC-16-53, August 8, 2016.

One particular concern [ ]^{a,c} is the influence of magnetic forces during dryout testing, potentially increasing the water gaps between the peripheral fuel rods and the channel walls, and hence improving dryout performance in those rods. Obviously Westinghouse has paid much attention to magnetic forces in the FRIGG loop. In the past, when using directly heated rods with the current running through the cladding, the magnetic forces were high which required additional support spacers in the test bundle. Material requirements did not allow the clad thickness to be reduced any further, which set an upper limit on the clad electrical resistance and hence a lower limit on the current through the rod (Magnetic Force ~ Current^2). When moving to indirectly heated rods, in which the current passes through an internal filament, the resistance could be increased significantly which allowed reducing the magnetic forces by a factor of ~4. Indirectly heated rods have been used for testing of all modern SVEA fuel designs. In addition, the magnetic forces for a 5x5 sub-bundle are significantly lower than for a full 10x10 bundle.

Westinghouse has calculated the magnetic forces affecting a SVEA-96 Optima2 test bundle under the most extreme heat conditions (at maximum critical power) and compared those to the capacity of the SVEA-96 Optima2 spacers against deformation. The results show forces always lower than the capacity of the spacer cells; hence the rod bundle will not contract when power is turned on. Also, the magnetic forces were confirmed to be too low to cause any significant rod bowing in-between the spacer levels.

Another concern [ ]^{a,c} is the potential inadequate positioning of thermocouples (TCs) with respect to the dryout locations observed at [ ]^{a,c} The actual orientations of the TCs during the FRIGG testing of SVEA-96 Optima2 fuel were carefully reviewed as part of the Root Cause Analysis. This is documented in [ ]^{a,c} [Reference A2-1, Appendix 2]. The results show that for the relevant rod locations and at the relevant axial levels, the TCs were indeed oriented in the direction where increased localized oxidation was observed in [ ]^{a,c}

AXIAL POWER SHAPE (APS) IMPACT – CPR ERROR ESTIMATED TO 0.20

The axial power shape impact actually belongs to the CP method but was singled out [ ]^{a,c} since it is well-known that CPR correlations based on the traditional annular length (AL) / boiling length (BL) concept used with critical quality correlations, such as the D4.1.5 CPR correlation used for SVEA-96 Optima2, have potential weaknesses in accounting for

differences in axial power shape (i.e. when extrapolating away from test loop conditions). **[** **]**^{a,c} This concept shortcoming was one of the reasons why Westinghouse developed the new, more phenomenological, D5 CPR correlation concept currently under NRC's review.

[

]^{a,c}

Westinghouse has compared predictions of the SVEA-96 Optima2 CPR correlation with higher order methods such as the D5 used for SVEA-96 OPTIMA3 and the sub-channel dryout prediction code, MEFISTO. The three methods agree within 0.05 of the predicted CPR for the specific conditions at **[** **]**^{a,c} The comparison is documented in Reference A2-1 in Appendix 2.

CHANNEL BOW – CPR ERROR ESTIMATED TO 0.15

Channel bow was an obvious candidate because it was the root cause of the steady-state dryout failure event of SVEA-64 fuel in the Oskarshamn 2 plant in 1988 (see discussion in Appendix 1). **[** **]**^{a,c} Westinghouse uses beta-quenched channels in **[** **]**^{a,c}, which are more resistant against bow induced by fast neutron fluence gradients, the dominating driver of channel bow for plants operating in 12 month cycles.

[**]**^{a,c} is very conscious of channel bow issues **[** **]**^{a,c} and are therefore monitoring the channel bow status of their core extensively every cycle. **[** **]**^{a,c} As a result, we have a clear picture of the potential influence of channel bow on CPR margins for freshly loaded fuel surrounded by older fuel assemblies

Westinghouse, together with **[** **]**^{a,c}, developed an improved CPR methodology to better account for the impact of channel bow on CPR margins during normal operation. An outline of this methodology is given in the paper "Dryout Methodology With New Steady-State Criterion and More-Accurate Statistical Treatment of Channel Bow" published in Nuclear Technology, Vol. 183 (Sept. 2013).

Additional evidence against any significant pin power distortions due to channel bow is obtained from the gamma scanning of the leaking assembly in **[** **]**^{a,c} as presented in Appendix 6. The gamma spectrometry results are documented in

[ ]a,c This report is considered Westinghouse internal documentation not typically referenced in topical reports. However, it can be made available for audit by NRC personnel.

Finally, we see dryout indications in several sub-bundles of the same fuel assembly (all four sub-bundles in **[ ]a,c**) which is contrary to any impact of a strong power tilt across the assembly. Thus, the CPR contribution of channel bow is essentially zero, when considering the occurrence of localized dryout in a large number of fuel assemblies at **[ ]a,c**

FLUX/SPECTRAL EFFECT – CPR ERROR ESTIMATED TO 0.10

The flux/spectral effects are not accounted for explicitly by the D4.1.5 CPR correlation, or historically by the CPR correlations from other vendors. Westinghouse explicitly accounts for these effects in the D5 correlation used for SVEA-96 OPTIMA3. In **[ ]a,c** [Reference A2-1, Appendix 2], as well as in several other studies performed by Westinghouse, it has been shown that this effect leads to, at most, a 0.02 difference in CPR for the CPR limiting assemblies. Also in this case, the argument of dryout indications in several sub-bundles of the same fuel assembly indicates the lack of any significant impact of the flux/spectral effect. Thus, the CPR contribution of flux/spectral effects is essentially zero when considering the occurrence of localized dryout in a large number of fuel assemblies at **[ ]a,c**

INLET LOSS – CPR ERROR ESTIMATED TO 0.05

The side entry inlet loss coefficient is discussed in detail in Sections 3.1 and 3.2. The CPR error was estimated by GNF to be 0.01 and corrected as part of SC02-15 [1]. According to GNF, the CPR error is in the range of 0.05 for SVEA-96 Optima2 fuel.

[ ]a,c detailed T/H characterization of the SVEA fuel inlet specifically for GE type reactors, i.e. using a relevant fuel support piece. **[ ]a,c** the detailed pressure drop contributions have been measured across the SVEA fuel inlet, starting from below the orifice inlet to a relevant fuel support piece and extending into the sub-assemblies. In this way, the pressure losses from each fuel and reactor component have been carefully evaluated to ensure the correct prediction of the fuel inlet pressure drop in the reactor.

[

]a,c

REFERENCES

A3-1 [

]a,c

APPENDIX 4: THE SVEA-CONCEPT INHERENT STABILITY

Westinghouse has extensively evaluated the SVEA sub-channel design with respect to unsteady and/or heterogeneous assembly inlet flow inducing (out-of-phase) flow oscillations between the SVEA sub-channels, and has demonstrated the inherent stability of the SVEA design.

In order to confirm the conclusion of these evaluations, Westinghouse initiated an independent review from a highly respected subject matter expert at the Royal Institute of Technology of Stockholm (KTH). This review is presented in [A4-1] and was provided to the NRC for information during the audit in March 2017. This appendix presents a summary of that report.

THE SVEA-DESIGN STABILITY CHARACTERISTICS

Two precautions were taken in the SVEA design to avoid the risk of parallel (sub-)channel instability. Pressure equalization along the entire channel by communication slots between sub-channels and significant inlet throttling to each sub-channel was obtained. The communication slots along the entire length of the fuel divert flow between the sub-channels and level out any pressure difference.

The stability characteristics of SVEA fuel were investigated experimentally, theoretically and during reactor operation. Both in-phase (single-channel) and out-of-phase (parallel sub-channel) instability were considered.

EXPERIMENTAL RESULTS

The SVEA design has been tested at the FRIGG loop stability test facility on several occasions; for the initial design, and each time a significant change to the design was introduced. These tests were performed using a full-scale mockup of the fuel and at realistic BWR core operating conditions covering a wide range of operation.

[

]a,c

For the full assembly tests, the power was increased up to the instability limit, but only if below the assembly critical power (dryout limit). For all tests performed at 70 bars, the assembly flow was so stable that the assembly power was limited by dryout, which always occurred first, rather than by flow oscillations of any significance. For the SVEA-64 test, the loop pressure had to be lowered to about 40 bars, where it became possible to induce flow instabilities, before dryout occurred.

Figure A4-1 summarizes the flow stability experience obtained from FRIGG loop testing of SVEA and 8x8 designs. The comparison indicates that the SVEA design has improved thermal-hydraulic stability compared to the open lattice 8x8 design.

It is worth noticing that the FRIGG stability results for SVEA-64 are consistent with experimental observations reported by other researchers studying flow instabilities in parallel channels with and without cross-connections.

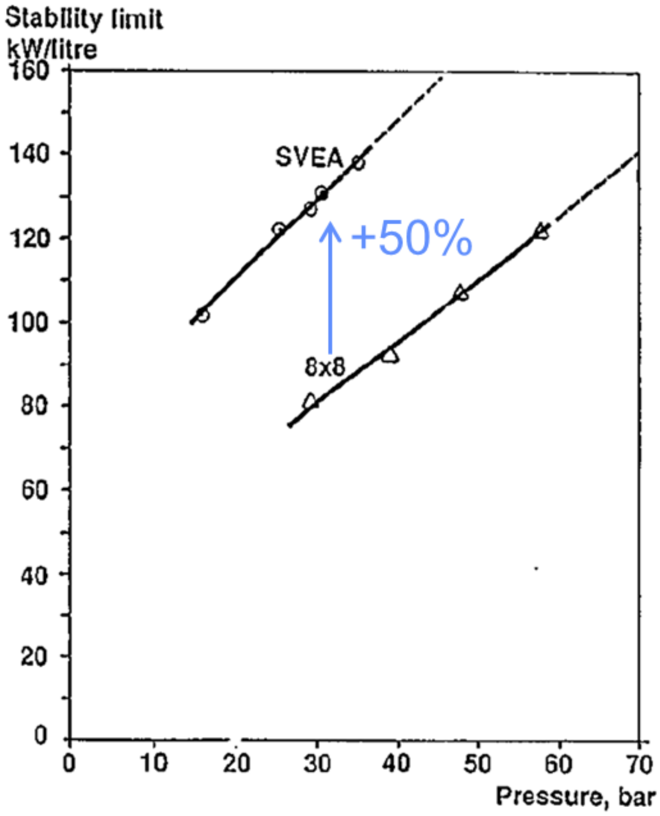
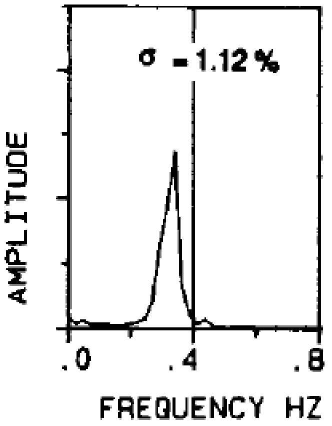


Figure A4-1 Stability limits at natural circulation for SVEA and 8x8 fuel.

Especially interesting are the tests performed for SVEA-64 fuel assembly to investigate the stability characteristics of the four sub-channels (a-d) at reduced pressures [A4-2]. Figure A4-2 shows the noise auto-spectra for the assembly inlet flow and the four sub-channels at natural circulation and 22 bars:



Noise auto-spectrum, assembly inlet flow

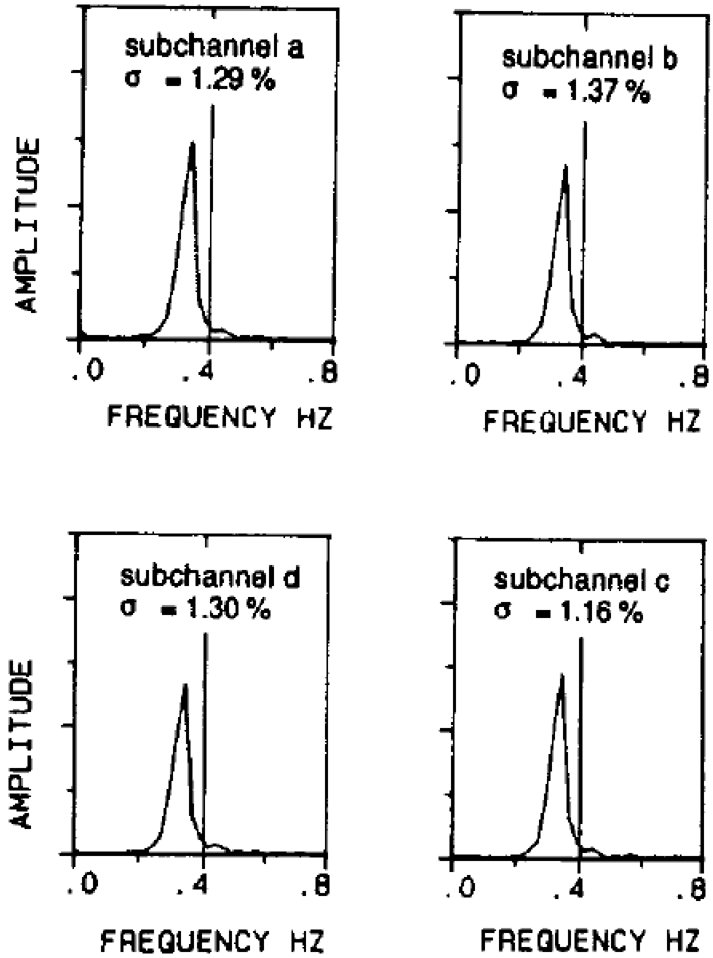


Figure A4-2: Noise auto-spectrum assembly inlet flow (top) and inlet flows to the four sub-channels (a – d)

The figure shows that the inlet flows to the four sub-channels fluctuate at the same characteristic frequency as the assembly inlet flow and have the same damping properties (decay ratios). This follows from the fact that the peaks appear at the same characteristic frequency and have the same spectral shapes (peak width ratios) as the assembly inlet flow.

These five flow signals have been cross analyzed in order to investigate the process couplings between adjacent sub-channels. Figure A4-3 presents the coherence results for the signals that refer to inlet flows to the pairs of sub-channels (a-b), (b-c), (c-d), and (d-a). Good signal coherence is present everywhere and in particular at the peak frequency observed in Figure A4-2.

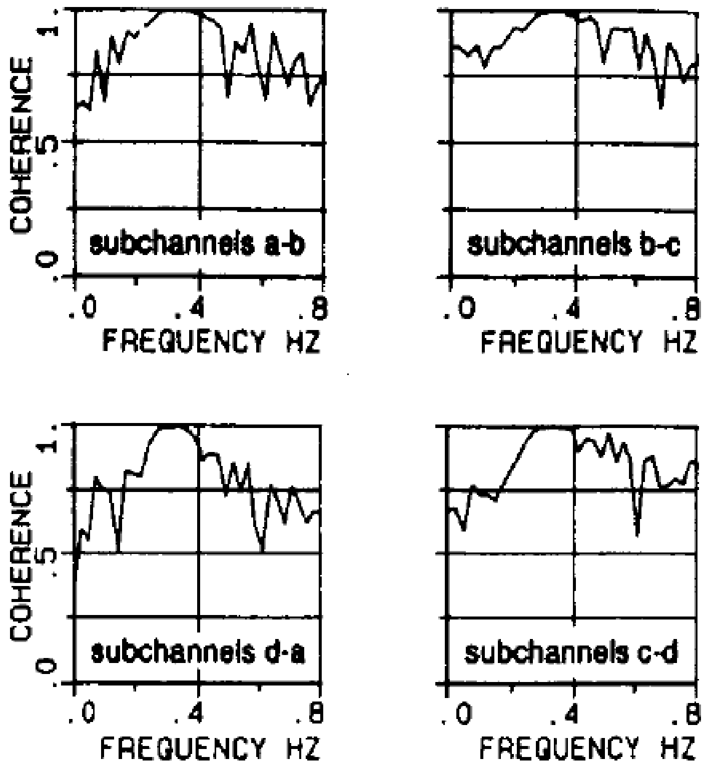


Figure A4-3: Coherence between inlet flows to adjacent sub-channels in SVEA-64 test

Figure A4-4 below, which presents the corresponding phase relations, shows that there are no significant phase differences between the signals. Together, Figures A4-2 and A4-3 demonstrate that the fluctuations occurring in the inlet flows to the four individual sub-channels are all strongly coupled to each other and are all in phase.

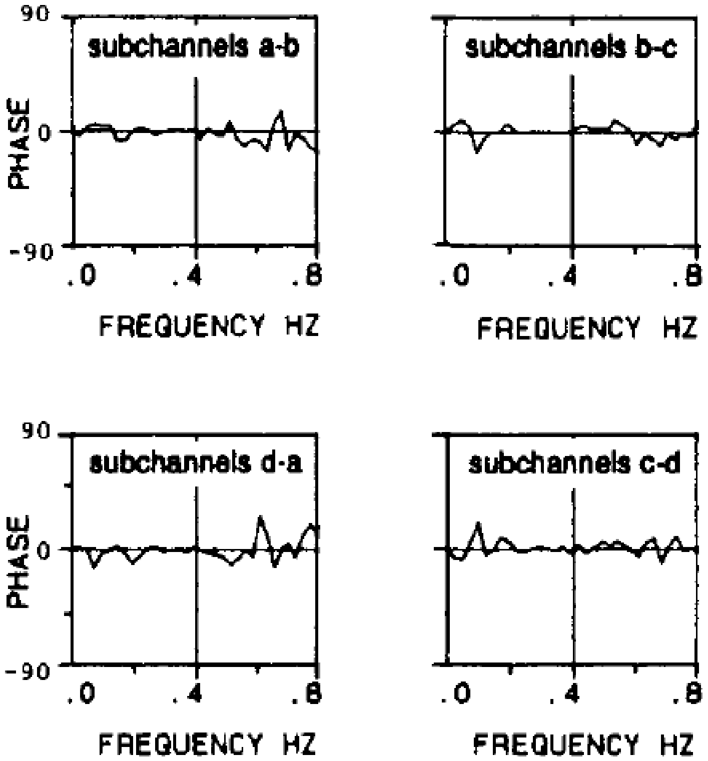


Figure A4-4 Phase relations between inlet flows to the adjoining sub-channels in SVEA-64

In summary, in the extensive FRIGG loop studies of the SVEA design always stable flow conditions were observed at operating pressures. Tests were also performed at reduced pressure (down to 22 bars) and in those tests it was demonstrated that

- the thermo-hydraulic fluctuations in the sub-channel inlet flows are always strongly coupled to each other and are always in phase with each other, and
- they represent the same characteristic fluctuations as in the assembly inlet flow.

OPERATING EXPERIENCE

In most BWRs the core coolant flow is measured via sensors that are located in the external parts of the recirculation loop. However, in-core SVEA stability behaviour can be properly assessed using in-core instrumentation provided in six BWRs of Westinghouse (ASEA-ATOM) design. Forsmark 1-3 and Oskarshamn 3 in Sweden and Olkiluoto 1 and 2 in Finland are provided with 8 flow-monitored fuel assembly positions as shown in Figure A4-5 below.

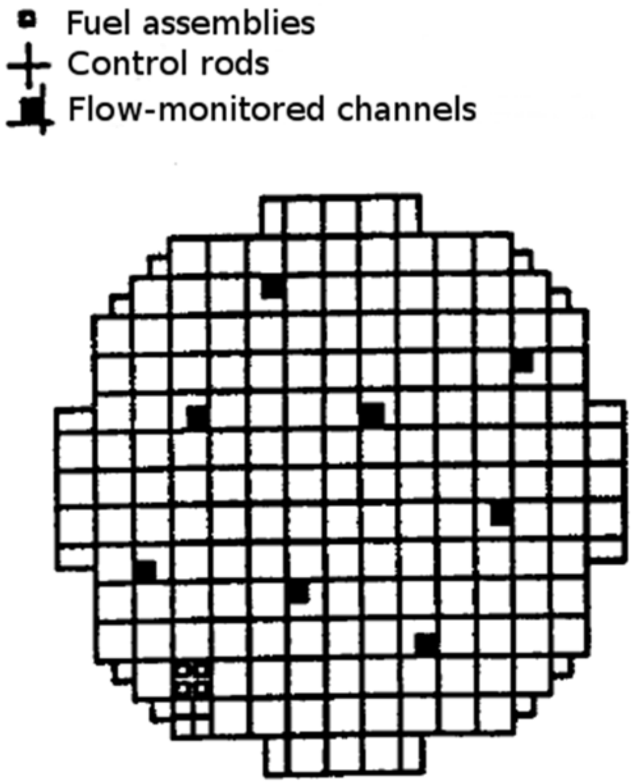


Figure A4-5 Flow-monitored positions in Oskarshamn 3 and Forsmark 3 reactors

The eight indicated fuel assembly positions contain instrumentation that allows a direct measurement of the inlet flow using pressure drop measurements over calibrated inlet orifice plates. The flow sensors have sufficiently fast response characteristics that time scales typical of flow instabilities can be measured.

A significant improvement in the thermal-hydraulic stability of SVEA fuel, as compared to 8x8 fuel, can be seen by comparing noise auto spectra for channel inlet flows to one high-power rated 8x8 lattice assembly and to an even more highly power rated SVEA-100 assembly, see Figure A4-6. Both recordings were made at 65% power and at relatively low core flow (close to the instability region in the power-flow map).

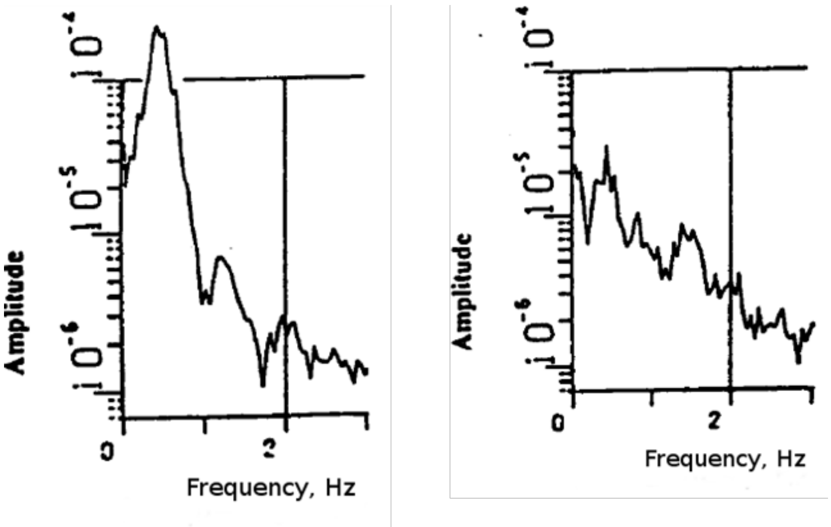


Figure A4-6 Noise auto-spectrum for channel inlet flow in 8x8 fuel (left) and SVEA-100 fuel (right) at 65% reactor power and relative assembly power 1.26 for the 8x8 fuel and 1.36 for SVEA-100.

The gradual transition from open lattice fuel to the SVEA design fuel has led to improved core stability in all plants. An example is shown in the table below for a plant in Germany. As can be seen in this table, increasing the fraction of SVEA fuel in the core significantly improves the decay ratio of the oscillations at similar operating conditions.

Table A4-1 Core stability characteristics for a core with 8x8 fuel, after first and second reload of SVEA fuel

	URK	Power	frequency	decay ratio
8x8	100%	46.3%	0.47 Hz	0.99
1:st SVEA-64	100%	47.7%	0.44 Hz	0.80
3:rd SVEA-64	99%	48.6%	0.45 Hz	0.52

THEORETICAL RESULTS

Both time-domain and frequency domain calculations confirm the solid stability properties of the SVEA fuel.

In time-domain calculations, POLCA-T was used to study the influence of cross-communications on the hydrodynamic stability of the entire SVEA fuel assembly. It was demonstrated that the communication slot area had to be reduced by a factor of more than 100 to notice any effect on the stability behavior of the assembly.

The frequency domain approach was applied to instability analysis of ventilated parallel boiling channels. The calculations were validated against experiments and they showed that the lateral ventilation (through communication slots between the sub-channels) significantly improves the stability characteristics of the system, and that parallel-channel instability can be excluded.

CONCLUSIONS

Stability characteristics of the SVEA fuel have been demonstrated experimentally and theoretically, as well as during reactor operation. All evidence indicates that SVEA fuel possesses robust stability properties. The main reasons for this behaviour are:

- Efficient ventilation between sub-assemblies, preventing out-of-phase oscillations and damping in-phase oscillations
- Adequate orificing at the inlet to the whole assembly
- Adequate orificing at the inlet to each sub-assembly
- Low local losses in the two-phase and exit parts of the fuel assembly

REFERENCES

A4-1 On SVEA Fuel Thermal-Hydraulic Stability – A Review Paper
Dr Professor Henryk Anglart, Royal Institute of Technology of Stockholm
Dated September 25, 2016

A4-2 Stability Investigations of SVEA-64 BWR Fuel
Paper presented at the Fourth International Topical Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-4),
Karlsruhe, F.R.G., October 10 - 13, 1989

APPENDIX 5: EXPERIMENTAL INVESTIGATIONS OF FLOW DISTRIBUTION IN SVEA FUEL

[ ]a,c
a flow mismatch at the inlet to each SVEA sub-bundle. [
]a,c

In spite of having been carefully addressed at the introduction of the SVEA channel design, the concern was further addressed by a new dedicated test program in the Westinghouse FRODE test loop.

The objective of the test was to determine if an inhomogeneous flow distribution at the fuel assembly inlet, potentially caused by the asymmetric location of the Fuel Support Piece (FSP) inlet orifice, could lead to an un-even flow distribution among the SVEA sub-channels, reducing the actual dryout margin.

FRODE LOOP HYDRAULIC TEST

A hydraulic test was performed at the FRODE loop to investigate the flow distribution between the SVEA four sub-assemblies. The purpose of the test was to investigate a potential asymmetry in the sub-assembly coolant flows caused by the asymmetric inlet flow conditions of the Fuel Support Piece (FSP) design at non-Westinghouse built BWRs. This inhomogeneous flow distribution could potentially reduce the margin to dryout.

Westinghouse performed a full-scale hydraulic test at the FRODE test loop in Västerås, Sweden, with a SVEA-96 Optima2 assembly mockup on a FSP with the four sub-assemblies instrumented. [

]a,c

The flow velocity (dynamic pressure) was measured with Pitot-tubes at the corner rod positions. These Pitot-tubes could be adjusted to different heights (axial positions). The following fuel inlet configurations were tested:

1. An Fuel Support Piece inlet section

2. A straight inlet section as reference

3. A straight inlet section with a known flow reduction to one of the sub-assemblies. The reduction was obtained by restricting the flow area through the sub-assembly bottom tie plate homogeneously by 30%.

The experimental setups are shown in figures A5-1 and A5-2 below.

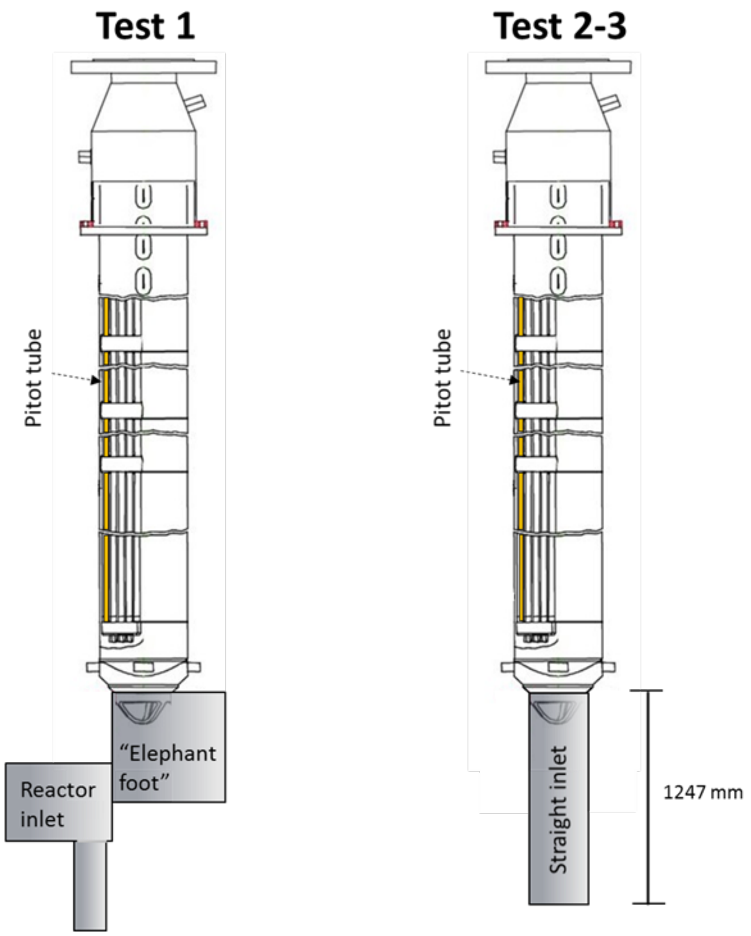


Figure A5-1 FRODE experimental setup showing the straight inlet section and the Fuel Support Piece "elephant foot" section.

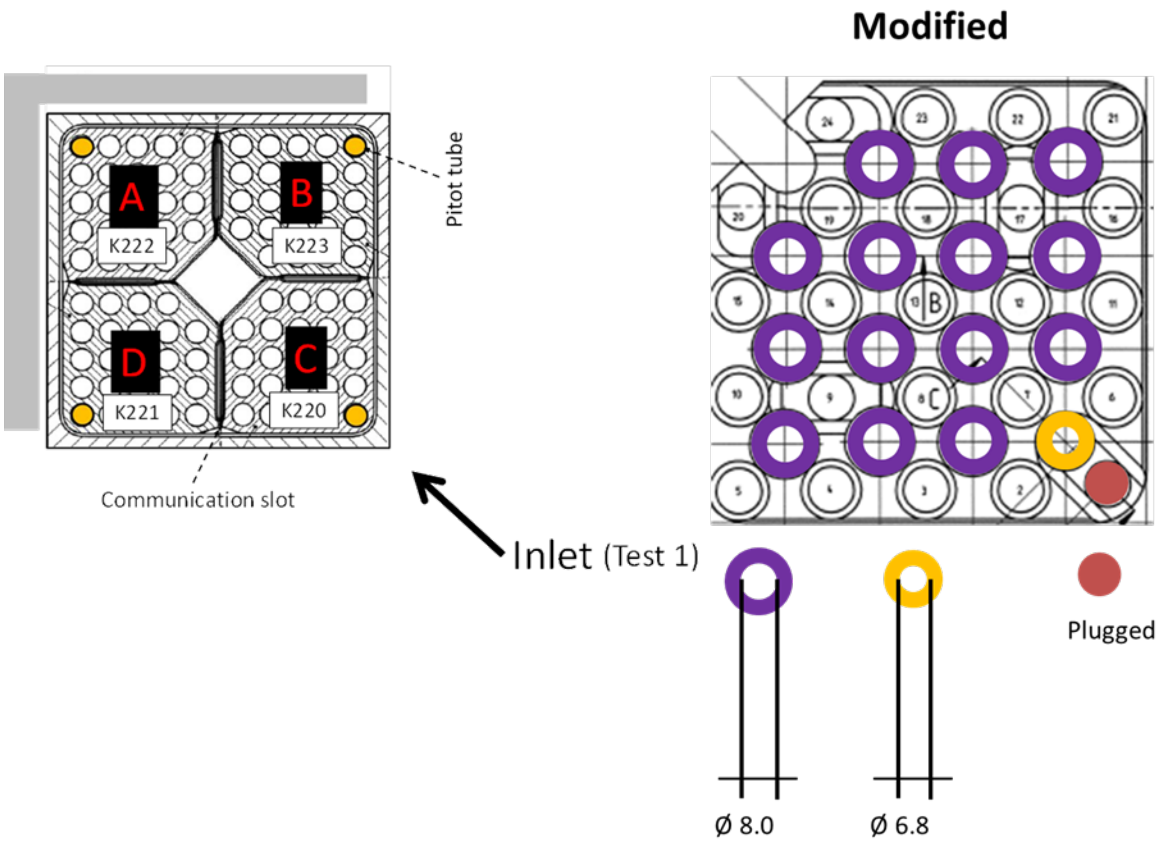


Figure A5-2 Fuel assembly inlet section for test 1 and 2 (left) and test 3 with a reduced inlet flow area (right)

Based on the flow velocities measured by Pitot tubes in each of the four sub-assemblies, the flow rate through sub-assembly A (i.e. the sub-assembly opposite to the elephant foot inlet orifice in Setup 1) was compared to the average flow rate in all four sub-assemblies. The relative flow rate in sub-assembly A, $(G_A - G_{avg}) / G_{avg}$, is plotted as function of the distance above the bottom tie plate in Figure A5-3, Figure A5-4 and Figure A5-5 for the different tests.

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In the test with the straight inlet, Figure A5-4, no significant flow differences were seen between any of the sub-assemblies, as expected since the entire setup is totally symmetric. This verifies the accuracy of the Pitot measurements to be within 1-2%.

In the test with the elephant foot inlet, Figure A5-3, lower than average flow was observed in sub-assembly A at measurement locations 10-15 cm above the tie plate. The flows through the other sub-assemblies did not show any significant differences. **[**
]a,c

In the test with the 30% flow area reduction, Figure A5-5, a larger flow mismatch was observed, **[** **]a,c** The flow differences even out very quickly over the first half meter and, in this case, become insignificant within 1 m.

The conclusions from the test results can be summarized as follow:

- The asymmetric geometry of the fuel support piece inlet gives rise to a minor reduction of the flow entering sub-assembly "A" in Figure A5-3 (positioned opposite to the orifice inlet). When compared to the sub-assembly average flow rate, the flow reduction is estimated at **[** **]a,c** at the entrance to the bottom tie plate.

- The flow mismatch is already reduced **[** **]a,c** elevation above the tie plate, and is completely evened out by the communication slots in the SVEA cross

wings when reaching 40 cm of the active fuel height (i.e. before entering 2-phase region).

- Even when reducing the flow area by ~30%, thus forcing a large flow mismatch between the sub-assemblies, the flow differences even out quickly over the first half meter and, in this case, become insignificant within 1 m.

- The accuracy of the above estimates is 1 - 2%.

It is concluded that the flow mismatch caused by different inlet types and obstructions quickly evens out between the sub-assemblies by means of the communication slots in the SVEA cross wings. In particular, the minor flow mismatch due to the elephant foot inlet evens out before entering the two-phase region of the fuel.

It should also be noted that the presence of significant and persisting inlet assembly flow heterogeneity would have caused a flow mismatch (and, thus, a burnup mismatch) among the SVEA sub-channels. Such burnup mismatch was not observed in the gamma spectrometry as described in Appendix 6.

The clear conclusion is therefore that the current assumption of an even flow distribution (prior to any redistribution due to sub-bundle power mismatch) in safety analysis for SVEA-96 Optima2 fuel, in **[ ]**[a,c] and any other plant with similar design of the fuel support piece, is appropriate.

APPENDIX 6: PIN BURNUP MEASUREMENTS AT [ ]a,c

As part of the root cause investigation, additional inspections were performed in June 2015 with the goal of measuring the integrated power generated by the leaking fuel assembly ([ ]a,c) during its first cycle of operation by using axial fission product distribution profiles. Gamma spectrometry was used and Cesium-137 profiles were compared to pre-calculated burnup profiles in order to identify potential deviations. **[**

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