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Thermal-Hydraulic Research Plan for Babcock and Wilcox Plants

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

R.Y. Lee

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Thermal-Hydraulic Research Plan for Babcock and Wilcox Plants

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R.Y. Lee

Division of Reactor and Plant Systems
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

ABSTRACT

This document presents a plan for thermal-hydraulic research for Babcock and Wilcox designed reactor systems. It describes the technical issues, regulatory needs, and the research necessary to address these needs. The plan also discusses the relationship between current and proposed research, and provides a tentative schedule to complete the required work.

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FOREWORD

This document presents a plan for thermal-hydraulic research for Babcock and Wilcox plants prepared by the Reactor and Plant Systems Branch, Division of Reactor and Plant Systems, in the Office of Nuclear Regulatory Research. This plan describes the technical issues, regulatory needs, and the research necessary to address these needs. The plan also discusses the relationship between current and proposed research, and provides a tentative schedule to complete the required work.

LIST OF ACRONYMS

AFW	Auxiliary feedwater
ANL	Argonne National Laboratory
ATOG	Anticipated transient operator guidelines
ATWS	Anticipated transients without scram
BCM	Boiler condenser mode
B&W	Babcock and Wilcox
ECCS	Emergency core cooling system
EPRI	Electric Power Research Institute
HPI	High pressure injection
HPIS	High pressure injection system
ICS	Integrated control system
INEL	Idaho National Engineering Laboratory
IST	Integral System Test
LOCA	Loss-of-coolant accident
LOMF	Loss of main feedwater
MIST	Multi-loop Integral System Test
NC	Natural circulation
NRC	Nuclear Regulatory Commission
OTSG	Once through steam generator
OTSG-FPO	Once through steam generator - full power operation
OTSG-LPSF	Once through steam generator - low power segment facility
PORV	Power operated relieve valve
PTS	Pressurized thermal shock
RCP	Reactor coolant pump
RELAP	Reactor Excursion and Leak Analysis Program
RVVV	Reactor vessel vent valve
SBLOCA	Small break loss-of-coolant accident
SG	Steam generator
SRI-2	Stanford Research Institute - 2
TSC	Technical Support Center
TAG	Technical advisory group
TMI-2	Three Mile Island - Unit 2
TRAC	Transient Reactor Analysis Code
UMCP	University of Maryland College Park
TSP	Tube support plate

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EXECUTIVE SUMMARY

This document provides a technical basis for planning and conducting current and future thermal-hydraulic experimental research for Babcock and Wilcox (B&W) designed reactor systems.

Following the Three Mile Island Unit 2 accident, NRC used best estimate thermal hydraulic computer codes such as RELAP5 and TRAC to analyze B&W reactors for small break loss-of-coolant accidents (SBLOCA). The lack of data to assess the validity of the calculated results of these codes led to the establishment of the Integral System Test (IST) Program in 1983. This program is jointly funded by the NRC, B&W, the B&W Owners Group, and the Electric Power Research Institute. The experimental data from the IST program is expected to be sufficient to assess code predictions of the integral response of B&W plants to SBLOCAs. Transients which have occurred in B&W plants since the establishment of the IST program, such as the loss of feedwater at Davis Besse and the loss of offsite power at Rancho Seco, indicate that unique conditions, not covered by the original IST program, can occur for such events. To address these issues, integral experimental data are therefore needed to validate the code predicted response for non-LOCA transients.

Best estimate codes have been and continue to be used by the NRC to address licensing issues and safety concerns such as decay heat removal capability and overcooling events which lead to pressurized thermal shock concerns, and in assisting the safety evaluation of abnormal transient operator guidelines. The large uncertainty in the code prediction of primary to secondary heat transfer and auxiliary feedwater spray distribution in the once through steam generator (OTSG) under a wide range of transients is due to limited or complete lack of information on OTSG behavior depending on the transient being analyzed. That is, a major part of the uncertainty in code prediction of integral system response is attributed to the OSTG modelling. Therefore, OTSG separate effects data as well as integral data on non-LOCA transients are necessary to quantify uncertainties in the code calculations and thereby aid in the confirmation of regulatory decisions.

The long-term resources needed for the experimental program are not yet committed. It is expected that several years will be required to provide an adequate data base for B&W reactors and the entire plan will not be completed without assistance from the industry. Programs and facilities are proposed in terms of their timeliness and cost effectiveness in producing the data to resolve technical issues. First, additional testing in the existing Multi-loop Integral System Test (MIST) facility will provide the integral test data for non-LOCA transients. The NRC codes are planned to be assessed against this data. Concurrently, separate effects thermal-hydraulic data for the OTSG will be obtained. The hydraulic aspects of auxiliary feedwater (AFW) in the once through steam generator (OTSG) are currently being characterized in the OTSG visual loop. It will be followed by the OTSG low power segment experiment to obtain the heat transfer coupling to the hydraulics for AFW. In addition, separate effect testing is also proposed to obtain thermal-hydraulic data on full power

OTSG performance for transients initiated from full power. These data will enable NRC to assess and improve OTSG modelling. A tentative schedule to obtain the required experimental data is also provided.

1. INTRODUCTION

The purpose of is : ument is to provide a technical basis for the current and proposed f: e thermal-hydraulic experimental research for existing Babcock and Wi (B&W) designed reactor systems.

Followi g the Three Mile Island Unit 2 (TMI-2) accident a number of regulatory issues were raised. In particular, it was required in Clarification of TMI Action Plan Requirements (NUREG-0737) Item II K.3.30 that small break loss-of-coolant accident (SBLOCA) calculational models be compared to applicable data. As a result of discussions with industry, a Test Advisory Group (TAG) was formed to make recommendations regarding the type of data base required for B&W plants to validate SBLOCA models. The Integral System Test (IST) Program jointly sponsored by the NRC, B&W, the B&W Owners Group and the Electric Power Research Institute (EPRI), was then formed in 1983 to acquire the required data. The primary experimental facilities in the IST program are the Multi-loop Integral System Test (MIST) facility, the University of Maryland College Park (UMCP) 2x4 loop, and the SRI-2 facility. Experimental data obtained in the IST Program are expected to be sufficient to provide the SBLOCA data base deemed necessary by NRC in response to NUREG-0737.

In addition to SBLOCA concerns resulting from TMI-2, more recent plant transients, such as the June 9, 1985 Davis-Besse and the December 26, 1985 Rancho Seco events, have reinforced the NRC staff's belief that B&W designed reactors exhibit complex behavior. In order to model this behavior in a best estimate manner in NRC codes, we need a better understanding of the detailed behavior. Some of the design features which contribute to the complex behavior of the primary side to certain off-normal secondary side transient conditions include:

(1) relatively small once through steam generator (OTSG) secondary liquid inventory

(2) auxiliary feedwater (AFW) which sprays directly on exposed tubes and may result in a rapid primary system cooldown

(3) reliance on a complex integrated control system (ICS) to automatically regulate and match steam and feedwater flow

(4) a low steam generator elevation in some plants which reduces the driving head available for natural circulation.

Therefore, an adequate data base is required to verify NRC best estimate codes, so they could be used to improve the NRC understanding about the overall plant thermal-hydraulic response to transients.

The remainder of this document addresses the need for additional experimental data in a B&W geometry to confirm the adequacy of these predictive code models. Section 2 summarizes the technical issues associated with B&W designed plants as well as the justification for obtaining the data. A description of existing

and proposed facilities which will provide needed data is given in Section 3. Section 4 contains information on the capability of each facility to provide data to resolve a particular technical issue and indicates whether the data will be used for code assessment or development. The program plan and a tentative schedule to obtain the necessary data for closure of the identified technical issues for B&W reactors is described in Section 5. Section 6 provides an overall summary while references are presented in Section 7.

2. REGULATORY NEEDS AND JUSTIFICATION

2.1 Primary Purpose

To aid in the staff understanding of the thermal-hydraulic response of commercial light water reactors during transients and off-normal operating conditions, the NRC frequently uses large complex computer codes such as RELAP5 and TRAC. The primary purpose of performing experimental transient simulations, both integral and separate effects, is to provide the needed data base for computer code assessment and development. Once a computer code has been assessed and calculation uncertainties determined, it can be used with a high degree of confidence to calculate similar transients in full-scale plants to assist in the understanding of the interactions among system components during off-normal operation.

2.2 Available or Currently Planned Data Base for B&W Reactors

Although some B&W plant transients data are available, they are primarily useful for qualitative assessment of thermal-hydraulic codes. This is due to the fact that measurements obtained from plant instrumentations are process instruments designed to provide operator information for operation of the plant. However, for quantitative assessment of the codes, the required experimental measurements far exceed those available from the plants in terms of quantity, frequency (i.e. sampling of parameter value) and types of measurements. Hence, it is necessary to obtain the appropriate experimental data to perform quantitative assessment of computer codes. In this respect, the experimental data base for B&W transient simulations is considerably smaller than the data base available for other plant designs. When the current IST program is completed, the data base will consist primarily of SBLOCA, feed and bleed and steam generator tube rupture tests. Experimental data is also needed to better validated the computer codes predictive capabilities over a wide range of transient initiating events such as steam line break, main feedline break, loss of main feedwater, steam generator overfill, risk dominant accident sequences (e.g. station blackout, SBLOCA without high pressure injection system (HPIS) available), and associated recovery strategies and procedures (e.g. single phase natural circulation cooldown with solid pressurizer).

2.3 Identification of B&W Design Related Thermal-Hydraulic Technical Issues

The integral data needed to assess the transient response of B&W reactors can be classified into different categories: SBLOCA, steam line break, main feedline break, loss of main feedwater, steam generator tube rupture, steam generator overfill, risk dominant accident sequences (e.g. station blackout, SBLOCA without HPIS available), natural circulation, along with associated recovery strategies and procedures. In addition, detailed understanding of the OTSG characteristics is required to improve the code modelling of the unique steam generator employed in the B&W design.

The following discussion on technical issues relating to B&W designs includes issues addressed by the MIST Technical Advisory Group and issues related to non-LOCA transients and risk dominant accident sequences (Ref. 1-4). These technical issues are summarized in Table 1. A brief synopsis follows on each technical issue, what experimental data are needed, and how that data will help resolve a particular technical issue.

Table 1 Summary of B&W technical issues

I. Small Break LOCA (Ref. 1)

   A. Break size
   B. ECCS operation
   C. Reactor coolant pump operation
   D. Break location
      1. Cold leg
      2. Hot leg - PORV
   E. Break isolation
   F. Reactor vessel vent valve (RVVV) operation
   G. Steam generator heat transfer

II. Steam Line Break (Ref. 2-4)

   A. Liquid carry over as a function of break size and liquid level
   B. Steam generator heat transfer
   C. Multiple failures
   D. Pressurized thermal shock

III. Main Feed Line Break (Ref. 2-4)

   A. Steam generator heat transfer
   B. Primary overpressure
   C. Break size

IV. Loss of Main Feedwater (Ref. 2,3)

   A. Steam generator heat transfer
   B. Auxiliary feedwater injection
   C. Multiple failures

V. Steam Generator Tube Rupture (Ref. 1-4)

   A. Overfill
   B. Number of tubes
   C. Break location
   D. Multiple failures
   E. Use of drains
   F. Fission product retention

Table 1 (continued)

VI. Steam Generator Overfill (Ref. 2-4)

A. Multiple failures
B. Water hammer
C. Steam generator heat transfer

VII. Risk Dominant Accident Sequences (Ref. 2,3)

A. SBLOCA w/o HPI
B. Station blackout
C. ATWS

VIII. Recovery Strategies and Procedures (Ref. 1-3)

A. Primary feed & bleed
   1. With reactor coolant pump operation
   2. Without reactor coolant pump operation
B. Single loop NC cooldown
C. Single-phase NC w/solid pressurizer
D. Abnormal transient operator guidelines (ATOG)

IX. Natural Circulation (Ref. 1-3)

A. Single-phase
B. Two-phase
C. Boiler condenser mode (BCM)
D. Steam generator driven instabilities
E. Cold leg oscillations
F. Interrupt/reestablishment
G. With high point vents
H. Non-condensible gas
I. RVVVs open

X. Once Through Steam Generator Characteristics (Ref. 4)

A. Auxiliary feedwater penetration
B. Tube wetting
C. Tube support plate flooding
D. Steam generator heat transfer
E. Condensation
F. Boiler condenser mode

2.3.1 Small Break LOCA

The technical issues with respect to the small break loss-of-coolant accident include the requirement to maintain coolant inventory in the core and to remove core decay heat. There are a large number of scenarios which can be postulated involving the SBLOCA. Controlling phenomena for the SBLOCA change depending on the size and location of the break, the operation of the reactor coolant pumps, assumptions regarding high pressure injection, low pressure injection, and accumulator injection, and whether or not the break can be isolated. The ability of the steam generator to provide decay heat removal capability at any point in the transient depends on the relative temperatures of the primary and secondary systems, the primary and secondary side inventories and AFW cooling.

Simulation of small break transients in integral facilities provides data for assessment of thermal-hydraulic codes. The small break transient is ideal for this assessment because of the large number of phenomena involved. The complex interaction of the controlling phenomena provides a test of the code capability to transition from model to model. Because of the large number of phenomena involved in a SBLOCA, it is possible to get the correct answer in a code calculation for the wrong reason. Consequently well qualified data which quantify phenomena in each component are necessary to provide an assessment base for the individual code models and the integrated system calculation.

2.3.2 Steam Line Break

Transient sequences involving excessive OTSG heat removal during a steamline break have been found to be significant contributors to risk for pressurized thermal shock (PTS) (Ref. 5). Major uncertainties exist in RELAP5 and TRAC computer calculations concerning OTSG heat removal during periods of AFW injection following blowdown of an OTSG. In these calculations, AFW injected at the top of the boiler tends to be bypassed over the top of the OTSG cylindrical shroud, into the steam annulus, and out the steam line to the break. Minimal downward penetration of AFW has been calculated by both RELAP5 and TRAC. If AFW actually penetrate further into the boiler, then the calculated OTSG heat removal would be greater, resulting in lower reactor coolant temperatures and a higher PTS risk. Both codes calculated reasonable AFW flow behavior based on one dimensional flooding considerations. There exists uncertainty in AFW flow behavior resulting from the possibility that such behavior is not one dimensional, but rather multidimensional. That is, during a steamline break, should the AFW penetrate downward around the periphery of the bundle and steam flows upward through the center of the bundle, it will result in more OTSG heat removal than has been calculated with the one dimensional modelling. Since multi-dimensional phenomena could not be excluded by available data along with the fact that the TRAC and RELAP5 calculated fluid temperatures for the main steam line break were significantly different, the NRC elected to make a very conservative assumption that the reactor vessel fluid was cooled to the saturation temperature of the affected steam generator. This was however much lower than the temperature calculated by either code. Thus, experimental data is required to define the uncertainty in code calculations for OTSG operation

during AFW injection. This should negate the necessity for NRC to make extremely conservative assumptions in its best estimate calculations.

2.3.3 Main Feedwater Line Break

Issues regarding main feedwater line breaks focus on core decay heat removal. A break in the main feedwater line results in a rapid loss of secondary side coolant inventory, a sudden reduction in primary to secondary heat transfer resulting in pressurization of the primary system. The timing of the events associated with a feedwater line break, the severity of the transient in terms of other potential failures, and the fuel rod cladding temperatures are dependent on the primary to secondary heat transfer. Integral systems data are needed to characterize the primary system response for different recovery scenarios. OTSG heat transfer data for this type of transient are needed to reduce code uncertainties in current code models and provide better predictive capabilities of primary side conditions.

2.3.4 Loss of Main Feedwater

The loss of main feedwater (LOMF) event is a common element in many sequences of interest. LOMF was a key factor in the June 9, 1985 transient at the Davis-Besse plant. Moreover, it can be a common initiator of accident sequences involving multiple failures such as a loss of all feedwater, station blackout, and anticipated transients without scram (ATWS).

When feedwater is lost, steam generator heat removal can be suddenly affected, typically causing a heatup and pressurization of the primary coolant system. Calculation of the thermal-hydraulic behavior within the steam generator during boiloff of the secondary coolant inventory is an important aspect of calculating plant response. Because of the small initial steam generator secondary coolant inventory, timing of event sequences and progression of the transient are significantly affected by unexpected variations in plant or systems behavior during this period.

Currently there is a lack of experimental data available to assess code uncertainty in the calculation of sequences involving LOMF combined with other failures for B&W geometry. As a result there are uncertainties involved in the analyses of event sequences and the timing of those events in probabilistic risk assessment studies. Operator actions played an important role in coping with the Davis Besse loss of all feedwater transient and prevented the transient from becoming more serious. However, all these actions were required to be performed in a very short time period. Timing is thus very important in determining risk for LOMF should these be combined with other failures. Additional experimental data are needed to better understand OTSG heat transfer and timing during LOMF events. These data will be used to assess code capabilities for calculating OTSG heat removal and may prove helpful in improving understanding of the human responses needed to cope with such transient initiated events.

2.3.5 Steam Generator Tube Rupture

A concern following a steam generator tube rupture is the release of fission products to the atmosphere. Primary system coolant is expelled through broken steam generator tubes into the steam generator secondary system; from there release to the atmosphere can occur through the steam generator safety relief valves. The human coping actions typically involve a reduction of primary coolant system pressure to minimize flow through the broken tubes. The degree of atmospheric release depends on the number of tubes broken and the effectiveness of plant emergency operating procedures that govern the human actions required.

The code prediction of the behavior of the straight-tube, low-inventory, counterflow heat exchanger, and the high emergency feedwater injection characteristics of the OTSG are known to be different from that of a U-tube steam generator following a steam generator tube rupture event and experimental data will serve to improve the current code predictive capability in regard to these differences.

Fission product retention is also expected to be importantly affected by the number of tubes broken, the break location, and will be governed by the human response actions taken to cope with the event. Experimental data for OTSG response during steam generator tube rupture events will serve to reduce uncertainty in the NRC's calculation of fission product releases and will provide a better assessment for the abnormal transient operating guidelines (ATOG).

2.3.6 Steam Generator Overfill

Steam generator overfill involves injection of main or emergency feedwater beyond the normal steam generator high level limit. This event typically results in overcooling of the primary coolant system, the main concern with respect to the PTS issue. Steam generator overfill, in the extreme, might result in a complete filling of the steam generator secondary systems. It is postulated that water hammer might occur in this situation as liquid enters the previously steam-filled steam lines with the turbine valves closed. Pressure excursions from water hammer might also have the capability of causing a main steam line break.

Steam generator overfill transients have occurred in operating plants in the past. Specific data required to improve code simulation of these events are the OTSG heat removal response and the likely conditions of fluid entering the steam lines especially if the AFW is on. This data will be used to assess code uncertainty in calculation of steam generator overfill response.

2.3.7 Risk Dominant Accident Sequences

The risk associated with a severe accident sequence is defined as the probability of the sequence leading to a specific consequence occurring times the consequence of the sequence. For a given reactor, most of the overall risk to the

public is associated with a few significant sequences usually called the risk dominant sequences. The risk dominant sequences have been identified from probabilistic risk assessment studies for B&W plants. These involve SBLOCA (possibly induced by other initiating events) without HPIS available, station blackout, and ATWS (Ref. 5). It is necessary that the computer codes be able to confidently predict the early thermal-hydraulic characteristics of these initiating events to sharpen the understanding of the associated sequence risks and those human responses best suited to cope with or terminate the intiating events.

The issues of concern include transient severity, timing of events, the effectiveness of potential recovery procedures, and the time by which the recovery procedures must be implemented in order to prevent core damage. Data addressing these issues will also assist in uncertainty determinations and code validation.

2.3.8 Recovery Strategies and Procedures

The issue related to recovery procedures is to obtain data to assist in the safety evaluation of ATOGs and as needed, their further optimization. These recovery procedures are of course important for B&W plants given the relatively small secondary coolant inventory in the OTSG and the inherent rapid response of this design.

The small secondary coolant inventory of the OTSG and the rapidity with which the OTSG can dry out have made it also desirable to obtain experimental data to examine alternate recovery strategies in the event that one or both OTSGs are unavailable. These alternate strategies might include cooling the plant with natural circulation utilizing a single OTSG, alternate emergency core cooling sources such as high point sprays, and primary system feed and bleed in the event that both OTSGs are unavailable. Issues related to primary system feed and bleed include the effect of reactor coolant pump operation, the effect of use of high point vents in conjunction with feed and bleed, and the optimal size of the high point vents. In addition, the transition from feed and bleed to a natural circulation cooldown using an OTSG is an issue because the ability to control the plant with a liquid solid pressurizer, which may result from feed and bleed, has not been adequately addressed by the predictive codes and thus can be improved by the experimental data.

2.3.9 Natural Circulation

Natural circulation is an important mode of heat removal during SBLOCA. Natural circulation provides a mechanism for removal of core decay heat without use of reactor coolant pumps. The concern is the ability of the operator to recognize from the existing plant instrumentation those conditions within the reactor system which may cause interruption of the natural circulation process. Natural circulation may be interrupted by voids in the hot leg U-bend, OTSG driven oscillations, and flow oscillations between the cold legs. Subsequently, the operator may be required to stabilize the cooling process and implement

means to restore natural circulation, such as using high point vents to remove steam and noncondensible gas from the hot leg U-bend. It should be noted that interruption of natural circulation does not imply that core damage is imminent.

Since the major modes of natural circulation in the B&W design include single-phase, two-phase, and boiler condenser mode (BCM), the codes should be assessed for their ability to accurately calculate the thermal hydraulic characteristics for all these modes. As such the affect of AFW on BCM cooling needs to be assessed. Unique design features of the B&W plants including the "candy cane" hot leg, the reactor vessel vent valves, and the vertical OTSG require that natural circulation data be obtained specifically for the B&W plant geometry. Such data is necessary to provide the data base for the validation of computer codes under natural circulation conditions.

2.3.10 Once Through Steam Generator Characteristics

Issues regarding OTSG characteristics deal with the ability of the AFW to remove decay heat during off-normal conditions. Identification and understanding of performance characteristics in the OTSG are as noted above, important, since the secondary side mass inventory is much smaller in the OTSG than in the U-tube steam generator.

Decay heat removal capability is provided by injection of AFW into the boiler region at the periphery and near the top of the tube bundle, and results in a complex counterflow situation within the boiler. Radial three dimensional penetration of the AFW into the tube bundle is restricted by the tube array and downward penetration is restricted by the tube support plates (TSPs) and the upward flow of steam.

Calculation of OTSG heat removal capability during AFW injection requires modeling of several phenomena including feedwater penetration, wetting of the tube walls, flooding at the TSPs, wall to fluid heat transfer, direct contact condensation on the secondary side, and condensation inside the tubes (boiler condenser mode). Many of these phenomena are encountered at different time during the course of any transient with AFW operation.

The code modelling of AFW behavior has been inferred from findings of steam water tests performed at a B&W plant at zero power and air-water AFW penetration testing under a limited range of expected steam generator thermal-hydraulic conditions. The validity of extrapolating the current methodology of steam generator modelling to a full-scale OTSG under a wide range of thermal-hydraulic conditions encountered during abnormal transients can be improved by the additional separate effects experimental data planned for herein.

2.4 Justification

Verified best-estimate computer codes are used by the NRC for a number of purposes. These include:

(a) Assisting in the assessment of operator guidelines and effectiveness of human responses to abnormal events.

(b) Confirming the safety margin in licensing and applicant analyses by performing auditing calculations.

(c) Examining perceived operational problems and/or plant design changes.

(d) Staff response to requests from the Commission, licensing boards and etc.

(e) Aiding in the understanding of operating reactor transient events whether within or external to the U. S.

For these purposes, it is necessary for these computer codes to predict plant response with reasonably accuracy and without artificial conservatisms being imposed for sake of margin, since sometimes these arbitrary conservatisms mask the important phenomena involved. The needed data will allow assessment of the NRC's best-estimate code capabilities for predicting B&W plant transients. In this manner, NRC will be able to use them to facilitate and enhance the soundness of NRC licensing decisions.

3. EXPERIMENT FACILITY DESCRIPTIONS

Experimental facilities which are currently available have been specifically designed to primarily address SBLOCA phenomena. NRC has received proposals (Ref. 6-8) to upgrade existing facilities as well as for the construction of new facilities to provide additional data for closure of the technical issues discussed in Section 2 which are not being addressed by the current IST program. These facilities could be divided into integral facilities and separate effects facilities, the latter addressing issues related to the performance of the OTSG.

3.1 Integral Facilities

Five integral facilities are identified in Table 2 as potential sources of experimental data for resolution of the technical issues shown in Table 1. These facilities are briefly described below and detailed descriptions are available in References 9, 10 and 11.

The MIST facility is located at the B&W Alliance Research Center in Alliance, Ohio, and is funded jointly by the NRC, B&W, B&W Owners Group and EPRI (Ref. 9). Table 2 summarizes the design parameters for the current B&W integral experimental facilities. MIST-III is the testing phase currently being conducted. The MIST-III configuration is designed to address primarily SBLOCA transients and natural circulation phenomena and is limited to decay power. MIST-IV is follow on to MIST-III which would include configuration changes to allow simulation of non-LOCA transients but would still be limited to decay power levels. MIST-V is a proposed follow on to MIST-III or MIST-IV which would include a facility upgrade to allow full scaled power simulation.

The UMCP facility (Ref. 10) and the SRI-2 facility (Ref. 11) are both in the facility checkout stages. They are both low pressure facilities with different geometric scaling ratios and were designed to address scaling atypicalities in the MIST facility in addition to providing code assessment data.

3.2 Separate Effects Facilities

There are several technical issues in Table 1 which relate specifically to detailed phenomena which occur in the OTSG. Technical issues associated with steam line breaks, feed line breaks and loss of feedwater transients require operation at full power. The 19 tube steam generators which are a part of the MIST facility could be utilized for investigation of OTSG behavior for these transients if a sufficient power supply was available. This facility is referred to as the OTSG-full power operation (OTSG-FPO). A proposal for a separate effects facility designed to specifically address the phenomena associated with auxiliary feedwater injection in an OTSG was recently submitted to NRC by INEL (Ref. 6). The proposed facility simulates a large segment of an OTSG which would allow investigation of TSP flooding, and steam generator tube wetting due to the multidimensional auxiliary feedwater penetration process. The proposal includes testing in air-water, and steam-water. The air-water OTSG

Table 2 Integral experimental facilities for B&W plant simulations

	MIST-III	MIST-IV	MIST-V	UMCP	SRI-2
Primary System Volume (ft^3)	20	20	20	21.2	6.98
Length Ratio	1	1	1	0.22	0.25
Heated Core Length (in)	144	144	144	24	32
Number Cold Legs	4	4	4	4	4
Number Hot Legs	2	2	2	2	2
No. SG Tubes/generator	19	19	19	28*	48
SG Tube Length (ft)	52.1	52.1	52.1	12.2	14
Core Power (MWt)	0.33	0.33	3.3	0.20	0.038
Max Operating Pressure (psi)	2500	2500	2500	300	100
Reactor Coolant Pumps	Yes	Yes	Yes	Yes	Yes
Status	Operating	Operating	Proposed	Operating	Operating

* The steam generator tube diameter in the UMPC facility is twice that of a B&W OTSG.

experiment is being conducted at INEL and testing was completed in FY 87. Based on the results of the air-water testing, the design for a low power steam-water multitubes OTSG experiment will be determined. This steam-water facility is referred to as the OTSG-low power segment facility (OTSG-LPSF).

4. CAPABILITY OF RESEARCH FACILITIES TO RESOLVE TECHNICAL ISSUES

This section describes the capability of existing or proposed experimental facilities to address and provide data for resolving B&W related technical issues. Table 3 lists the issues from Table 1 and indicates which of the experimental facilities could be used to address the technical issue. The current IST program centers around SBLOCA tests and assessment of the codes for SBLOCA transients since this was the major technical issue being addressed at the time. The MIST-III, UMCP, and SRI-2 facilities described in the previous section form the experimental basis for the IST program. In the IST program tests, particular attention is given to conditions under which natural circulation cooling would be lost due to loss of liquid inventory breaking the path to the steam generator, cooling by condensation of steam in the steam generator tubes or boiler-condenser mode (BCM) cooling, and conditions under which natural circulation cooling would be reestablished. Testing of primary cooling by feed and bleed is also included, as well as testing of steam generator tube ruptures.

In addition to directly providing transient data, the UMCP and SRI-2 facilities will provide insight into the effect of facility atypicalities on the understanding and interpretation of the MIST-III test data.

The MIST-III facility has several limitations which prevent adequate simulation of certain phenomena beyond the small break data needs identified by the TAG. Break sizes are limited to small breaks by the currently installed leak locations and the catch tanks. Power is limited to decay heat levels by the available power distribution system. In addition, the heat rejection of the secondary side is also limited to decay heat levels by the present condenser system. This also precludes performing steam generator feedline and steam line breaks due to a lack of break flow condensing capability. The design of the facility is one-dimensional (tall and thin) and thus multidimensional phenomena, such as auxiliary feedwater spray distribution, cannot be properly simulated.

To overcome the facility limitations in MIST-III, and address technical issues not covered by the current IST Program, several new facilities or modifications to existing facilities have been proposed.

These facilities, as described in Section 3, include MIST-IV, MIST-V, OTSG low pressure segment facility, and the OTSG-FPO. Table 3 summarizes the ability of each facility to provide data for technical issue resolution. The ability of the UMCP 2x4 loop and SRI-2 facilities to provide data beyond that of SBLOCA and natural circulation will best be determined after the examination of the composite data from these facilities.

Table 3 Experimental facilities for resolution of B&W technical issues

Technical Issue	Code Assessment					Code Development	
	MIST III	MIST IV	MIST V	UMCP	SRI-2	OTSG FPO	OTSG LPSF
I. Small break LOCA							
A. Break size	**		*	**	**		
B. ECCS operation	**		*	**	**		
C. RCP operation	**		*	**	**		
D. Break location							
1. Cold leg	**		*	**	**		
2. Hot leg - PORV	**		*	**	**		
E. Break isolation	**		*	**	**		
F. Reactor vessel vent valve (RVVV) operation	**		*	**	**		
G. Steam generator (SG) heat transfer	*'		*	*'	*'		

The small break LOCA transients have, for the most part, been addressed by the MIST-III program with experiments performed in the UMCP and SRI-2 facilities to address atypicalities in the MIST facility. MIST-V will provide data in simulating the impact of the initial core power levels and flow characteristics on the post-SBLOCA transient.

* : capable
*': low power simulation only
** : currently being addressed

Table 3 (continued)

Technical Issue	Code Assessment					Code Development	
	MIST III	MIST IV	MIST V	UMCP	SRI-2	OTSG FPO	OTSG LPSF
II. Steam line break							
A. Liquid carry over as a function of break size and liquid level			*			*	
B. SG heat transfer		*'	*			*	*'
C. Multiple failures			*			*	
D. Pressurized thermal shock			*				*

A facility capable of simulating the full power initial condition in the primary side is needed to address the issues surrounding the steam line break transients such as effect of break size on the system response, amount of liquid carryover, and liquid level response. Multiple failures associated with the steam line break include steam generator tube rupture. The MIST-V facility could be used to obtain the required system data. Liquid carry over is expected to be better simulated in a full height facility due to the higher local vapor velocity. Therefore, phase separation in SG secondary will be more prototypic in a full height facility. The most severe cooldown condition with regard to the pressurized thermal shock (PTS) issue has been shown to occur if a main steam line break occurs during hot standby conditions. The separate effects OTSG-FPO facility would provide data for steam generator heat transfer.

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* : capable
*': low power simulation only
** : currently being addressed

Table 3 (continued)

Technical Issue	Code Assessment					Code Development	
	MIST III	MIST IV	MIST V	UMCP	SRI-2	OTSG FPO	OTSG LPSF
III. Main feed line break							
A. SG heat transfer		*'	*			*	
B. Primary overpressure			*				
C. Break size			*			*	

Proper simulation of the system response during a main feedline break such as system overpressurization due to loss of cooling capability and heat transfer during high power boil-off require use of scaled power for simulation. This power simulation requirement necessitates the use of the MIST-V facility in order to obtain system response data. However, atypical mass to inventory distribution in MIST-V will distort the primary overpressurization. Steam generator separate effects heat transfer data can be obtained in the OTSG-FPO facility.

* : capable
*': low power simulation only
** : currently being addressed

Table 3 (continued)

Technical Issue	Code Assessment					Code Development	
	MIST III	MIST IV	MIST V	UMCP	SRI-2	OTSG FPO	OTSG LPSF
IV. Loss of main feedwater							
A. SG heat transfer		*'	*			*	
B. AFW injection			*				*
C. Multiple failures			*				

The primary system response and the steam generator response during at the elevated high power portion of a loss of main feedwater transient prior to reactor trip is important for understanding this transient. The loss of feedwater transient is also a common initiating event for many of the multiple failure sequences including total loss of feedwater, ATWS, and station blackout. The power simulation requirement necessitates the use of the MIST-V facility for proper simulation of the initial part of this LOMF transient. However, for the latter part of the transient, the simulation of the multidimensional behavior of OTSG thermal hydraulics due to the auxiliary feedwater is needed. The OTSG-FPO facility can be used for obtaining specific heat transfer data associated with the loss of feedwater transient.

* : capable
*' : low power simulation only
** : currently being addressed

Table 3 (continued)

Technical Issue	Code Assessment					Code Development	
	MIST III	MIST IV	MIST V	UMCP	SRI-2	OTSG FPO	OTSG LPSF
V. Steam generator tube rupture							
A. Overfill			*				
B. Number of tubes	**		*				
C. Break location	**		*				
D. Multiple failures	**		*				
E. Use of drains			*				
F. Fission product retention			*				

Most of the technical issues associated with steam generator tube rupture transients occur after plant trip and require that only decay power. Many of the transients were simulated in the MIST-III testing. Additional experiments in the MIST-IV or MIST-V could be perfomed to examine different recovery strategies.

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* : capable
*' : low power simulation only
** : currently being addressed

Table 3 (continued)

Technical Issue	Code Assessment					Code Development	
	MIST III	MIST IV	MIST V	UMCP	SRI-2	OTSG FPO	OTSG LPSF
VI. Steam generator overfill							
A. Multiple failures			*				
B. Water hammer			*				
C. SG heat transfer		*'	*			*	*

For some steam generator overfill transients, the phenomena of concern may occur prior to a reactor trip. The temperature of the liquid entering the steam line will depend on the main feedwater flow rate and core power history. Where the full power simulation is required, MIST-V could provide the data. OTSG-FPO will be able to provide separate effects steam generator data. To address those steam generator heat transfer conditions which occur later in the transient, MIST-IV could be utilized.

* : capable
*': low power simulation only
** : currently being addressed

Table 3 (continued)

Technical Issue	Code Assessment					Code Development	
	MIST III	MIST IV	MIST V	UMCP	SRI-2	OTSG FPO	OTSG LPSF
VII. Risk dominant accident sequences							
A. SBLOCA w/o HPI		*	*				
B. Station blackout		*	*				
C. ATWS			?				

The small break LOCA (without HPI) as well as the station blackout sequence, can be simulated in the MIST-IV facility because of these events feature on early reactor trip. For ATWS, the power level level simulation with feedback from core fluid temperature, will only provide boundary conditions for thermal-hydraulic phenomena, it cannot, of course, be expected to provide data for assessment of code kinetics models.

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* : capable
*' : low power simulation only
** : currently being addressed
? : limited capability

Table 3 (continued)

Technical Issue	Code Assessment					Code Development	
	MIST III	MIST IV	MIST V	UMCP	SRI-2	OTSG FPO	OTSG LPSF
VIII. Recovery strategies and procedures							
A. Primary feed and bleed							
1. With RCP operation	**		*				
2. Without RCP operation	**		*				
B. Single loop NC cooldown		*	*				
C. Single-phase NC w/solid pressurizer		*	*				
D. ATOG			*				

Since ECC injection does not occur until after reactor trip, the phenomena of interest could occur at low plant power. Consequently, all of the recovery strategies and procedures listed above can be simulated using the MIST-IV facility with appropriate modifications. The MIST-V is capable of simulating all of the above recovery strategies and procedures.

* : capable
*': low power simulation only
** : currently being addressed

Table 3 (continued)

Technical Issue	Code Assessment					Code Development	
	MIST III	MIST IV	MIST V	UMCP	SRI-2	OTSG FPO	OTSG LPSF
IX. Natural circulation							
A. Single-phase	**		*	**	**		
B. Two-phase	**		*	**	**		
C. Boiler condenser mode	**		*	**	**		
D. SG driven instabilities	**		*	**	**		
E. Cold leg oscillations	**		*	**	**		
F. Interrupt/reestablishment	**		*	**	**		
G. With high point vents	**		*				
H. Noncondensible gas	**		*				
I. RVVVs open	**		*	**	**		

The MIST-III test series was specifically designed for detailed investigation of natural circulation phenomena especially in conjunction with small break LOCAs. Natural circulation was also the main area of investigation in the UMCP facility and was also emphasized in the SRI-2 facility.

* : capable
*': low power simulation only
** : currently being addressed

Table 3 (continued)

Technical Issue	Code Assessment					Code Development	
	MIST III	MIST IV	MIST V	UMCP	SRI-2	OTSG FPO	OTSG LPSF
X. OTSG characteristics							
A. Auxiliary feedwater penetration							*
B. Tube wetting							*
C. TSP flooding							*
D. SG heat transfer						*	*
E. Condensation							*
F. Boiler condenser mode							*

Because of the steam generator performance is significant in most of the technical issues addressed in this document, it is important to understand the local phenomena which occur during these transients. The obtaining of detailed local phenomena implies the performance of separate effects testing. The steam generator technical issues fall into two categories. The first involves the heat transfer, entrainment, and carryover during steam line breaks and requires simulating full power operation. These data needs can be addressed in the OTSG-FPO facility. The other data needs occur during auxiliary feedwater injection where the key issues involve the penetration of the auxiliary feedwater into the tube bundle interior and its progression downward to the tube support plate. The OTSG-LPSF is specifically designed to address these multidimensional issues.

* : capable
*': low power simulation only
** : currently being addressed

Table 3 (continued)

Key to facilities identified in Table 3

MIST-III: This is the MIST experimental program presently underway at B&W and is limited to those experiments outlined under the current testing agreement.

MIST-IV: This proposed facility includes upgrades to the MIST-III facility to provide additional testing capability but will be limited to scaled decay power levels.

MIST-V: This proposed facility is the MIST-IV facility upgraded to provide full scaled power capabilities.

UMCP: The University of Maryland, College Park facility is a low pressure, reduced height facility operated by the University of Maryland to address atypicalities in the MIST facility and to obtain data for code assessment. Power is limited to decay heat levels.

SRI-2: The Stanford Research Institute-2 facility is a low pressure, reduced height facility having decay power levels and was also designed to support the MIST-III testing program.

OTSG-FPO: The proposed Once Through Steam Generator-Full Power Operation facility utilizes one of the 19 tube steam generators currently used in the MIST program in a simulated full power, separate effects testing mode.

OTSG-LPSF: The proposed Once Through Steam Generator-Low Power Segment Facility is designed to investigate OTSG thermal-hydraulic behavior during auxiliary feedwater injection.

5. PROGRAM PLAN

The rationale used in developing the thermal-hydraulic research plan for B&W plants will be discussed here. In formulating the research plan, two major considerations were established to determine the experimental facilities to be used to provide additional thermal-hydraulic data for B&W plants. The two major considerations were (1) timeliness, and (2) cost effectiveness in obtaining the data. The specific considerations used were (1) the timely resolution of data needs by a particular experimental program or analysis, (2) the cost effectiveness of conducting additional work in existing facilities, (3) retaining expertise and continuity of staff, (4) minimizing cost in integration of experimental results into all phases of analyses, and (5) cost effectiveness in conducting several experiments at one site thereby utilizing the common personnel skills and equipment. For completeness, support programs that are vital to the success of the thermal-hydraulic research plan for B&W plants are also included. These support programs are (1) the analysis work on pre- and post-test analyses for MIST, (2) the INEL Thermal Hydraulic Technical Support Center, and (3) the Argonne National Laboratory (ANL) modelling of two-phase flow. Table 4 is a list of the facilities and support programs. The following is a brief discussion of the importance of each of these projects to NRC in resolving all the technical issues previously identified in Table 3.

Table 4 Experimental facilities and support programs

1. Thermal-Hydraulic Technical Support Center (TSC)
2. OTSG Visual Loop
3. MIST Analysis
4. MIST-IV
5. OTSG Full Power Operation (OTSG-FPO)
6. OTSG Low Power Segment Facility (OTGS-LPSF)
7. UMCP 2x4 Loop
8. ANL Modelling of Two-Phase Flow
9. MIST-V

5.1 Thermal-Hydraulic Technical Support Center

The NRC has undertaken an initiative to better integrate and synthesize the results of safety research into the regulatory process. The Thermal-Hydraulic TSC at INEL is a part of that initiative. This need is primarily driven by the fact that some of the more significant thermal-hydraulic transients in operating nuclear plants have not been limited to single failure of a system or component, but have involved multiple failures and/or human errors. This experience has highlighted the need for a multidisciplinary view in understanding the risk of transients which span several disciplines (e.g., thermal hydraulics, risk assessment, human performance, structure and equipment reliability). Although the thermal-hydraulic TSC does focus on providing the technical basis for resolving thermal-hydraulic issues, it can draw on other disciplines, as needed, to aid in this resolution.

5.2 OTSG Visual Loop

The knowledge of the effects of AFW penetration and flooding on tube support plates in an OTSG is rather limited. The available data are from steam-water tests performed on a full-scale OTSG, and air-water AFW penetration testing under a limited range of expected thermal-hydraulic conditions. Based on existing experimental data, the current code modeling practice employed by NRC is to fix the heat transfer area wetted by AFW at a constant percentage of the full total area regardless of the type of transient. Hence, the extrapolation of current modeling techniques to full-scale commercial plant transient analyses entails a large degree of uncertainty in the calculated primary to secondary heat transfer. Therefore, experimental data are required on the effect of AFW on the thermal-hydraulic behavior in an OTSG with a large number of tubes. The visual OTSG air-water loop will provide the hydrodynamic data on wetting of tube walls and flooding at tube support plates by AFW. In addition, the results will provide insight for the design of the low power OTSG separate effects test (OTSG-LPSF).

5.3 MIST Analysis and MIST-IV

The current MIST-III program is designed primarily to address SBLOCAs in a B&W plant. The current plan for MIST-IV is to conduct a limited number of tests to obtain a minimum set of integral experimental data to benchmark best estimate codes for non-LOCA transients at low power conditions. These include risk dominant accident sequences (e.g. station blackout, SBLOCA without HPI), and cooldown strategy (e.g. steam generator tube rupture, both loop interruption, HPI-PORV cooling). From the standpoint of continued use of an existing facility and retaining expertise within a program, performing the MIST-IV work shortly after the conclusion of MIST-III was the next logical step to take. This has minimized the start up time and cost for MIST-IV.

5.4 OTSG Full Power Operation

MIST-IV is capable of providing data on secondary side behavior and the associated effects on the primary side response to a feedwater line break, a SG overfill transient and the loss of feedwater transient at a simulated decay power level. However, data is needed on SG heat transfer for these transients as well as for steam line break at full scaled power. Since multidimensional behavior is not expected to be important for these transients, a 19 tube full length OTSG will suffice for obtaining the necessary information. The full power simulation on the primary side of the OTSG could be provided by an adequate steam power system.

5.5 OTSG-LPSF

The OTSG-LPSF is designed to obtain thermal-hydraulic data for auxiliary feedwater behavior in an OTSG under a wide range of thermal-hydraulic conditions. However, based on the results of the visual air-water testing, the need

for this experiment will be determined.

5.6 UMCP 2x4 Loop

The UMCP 2x4 Loop is a low pressure experimental facility simulating a B&W plant of the lowered-loop design. The facility will provide data to complement the MIST data base by addressing the effects of some of the MIST design atypicalities and the effects of scale distortion inherent in scaled experiments. Natural circulation, SBLOCA, steam generator tube rupture, and feed and bleed, are the major transients to be conducted at this facility. The code will be assessed against the integral data from this facility which is designed on a different scaling approach relative to other integral facilities.

5.7 ANL Modelling of Two-Phase Flow

This project provided a visual air-water experimental loop that simulated the hot leg U-Bend of a lowered-loop B&W plant to study the thermal-hydraulic phenomena associated with SBLOCA. The phenomena studied were two-phase natural circulation, U-Bend flow separation, flow termination and flow resumption. Freon-113 will also be used as a working fluid to study the additional influence of the evaporation and condensation effect which the air-water experiment could not provide. The data from this program is used to assess code predictions of phenomena in the hot leg U-bend during various transients. In addition, the program also provides scaling evaluation of various experimental facilities, such as the UMCP 2x4 Loop, and MIST.

5.8 MIST-V

From Table 3, one recognizes that MIST-V is capable of providing comparable data from facilities such as MIST-IV and OTSG-FPO. The key advantage of MIST-V is the ability to provide integral system test data to benchmark codes for non-LOCA transients. However, the NRC staff felt that the separate effects programs such as the OTSG Visual, OTSG-LPSF, OTSG-FPO combined with a limited number of integral effects tests in MIST-IV will provide the necessary data to resolve outstanding technical issues.

5.9 Tentative Schedule

A tentative schedule to obtain the necessary experimental data is shown in Table 5. The MIST-IV program was initiated in FY 87 immediately following the MIST-III program and would be completed by the early part of FY 89. MIST analysis activities would be completed by FY 90. The OTSG visual experiment was completed in FY 87. Commencing in FY 88, NRC and the industry are jointly investigating technical issues and data needs to further the understanding of OTSG behavior under steady state, transient and accident conditions. The investigation will identify the need for additional thermal-hydraulic data for OTSG. Tentatively, the OTSG-FPO and the OTSG-LPSF are scheduled for completion in FY 90. According to this schedule, all MIST and OTSG related activities would be completed by the end of FY 90. These schedules are judged to provide an acceptable timeframe to ensure that the currently identified B&W technical issues will be addressed by independent experimental data and validated predictive codes.

Table 5 Tentative schedule

Schedule	FY 87 FY 88 FY 89 FY 90
MIST-IV	xxxxxxxxxxxxxxxxxxxxx
MIST-V or OTSG-FPO	xxxxxxxxxxxxxxxxxxxxxxxxxxxx
MIST Analysis	xxxxxxxxxxxxxxxxxxxxxxxxxxxx
OTSG Visual Experiment	xxxxx
OTSG-LPSF	xxxxxxxxxxxxxxxxxxxxxxxxxxxx
TSC	xxxxxxxxxxxxxxxxxxxxxxxxxxxx
UMCP 2x4 Loop	xxxxxxxxxxxxxxxxxxxxxxxxxxxx
ANL	xxxxxxxxxxxxxxxxxxxxx

6. SUMMARY

The best estimate codes used to simulate B&W power plant behavior have been and are being used by NRC to address licensing issues and safety concerns such as decay heat removal capability, overcooling events which lead to pressurized thermal shock concerns, and in assisting the safety evaluation of the operator guidelines to recover from abnormal transient events. The uncertainty in the code's capability to accurately predict phenomena, such as primary to secondary heat transfer, auxiliary feedwater spray distribution in the steam generators, and boiler condenser mode steam generator heat transfer, constrains the usefulness of best estimate codes in the resolution of licensing issues and safety matters.

Compared to other PWR designs, there is a more limited B&W experimental data base available to assess and validate the codes and reduce uncertainties in calculated results. Currently, the IST program primarily addresses the SBLOCA and natural circulation issues for B&W plants. Technical issues dealing with a wide range of other transients such as feedwater line breaks and risk dominant accident sequences (e.g., station blackout, SBLOCA without HPI) creates a further need for facilities to produce relevant data. Programs and facilities proposed to produce this data include MIST-IV, the OTSG visual, the OTSG-LPSF, and the OTSG-FPO.

With the objective of completing this experimental data program in a timely manner, a tentative schedule for the experimental and support programs was developed. Data will be available by the end of FY 90 to address the identified B&W technical issues. This timeframe is consistent with the identified data needs for code assessments.

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13 ABSTRACT (200 words or less)

This document presents a plan for thermal-hydraulic research for Babcock and Wilcox designed reactor systems. It describes the technical issues, regulatory needs, and the research necessary to address these needs. The plan also discusses the relationship between current and proposed research, and provides a tentative schedule to complete the required work.

14 DOCUMENT ANALYSIS - a KEYWORDS/DESCRIPTORS	15 AVAILABILITY STATEMENT
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