INEL
Idaho National Engineering Laboratory

52-003

July 21, 1992

Dr. L. M. Shotkin
U. S. Nuclear Regulatory Commission
5640 Nicholson Lane
MS NL/N-353
Washington, DC 20555

COMMENTS ON DR. ZUBER REPORT TO DR. CATTON ON ROSA-IV/AP600 MEETING, JUNE 3-4, FROM JUNE 23, 1992 - SMM-38-92

Dear Dr. Shotkin:

I have read Dr. Zuber's report to Dr. Catton from June 23, 1992. Upon reading his comments, I have the impression he discusses application of the ROSA-IV test facility for demonstration type tests or AP600 simulation rather than for generation of data for code assessment. His argumentation of significance of distortions focuses on the capabilities of ROSA-IV to duplicate the AP600 behavior. Irregardless of ROSA-IV's shortcomings, duplication of reference system behavior in scaled facilities under two-phase transient flow conditions is a priori impossible. On page 13 of his memorandum, Dr. Zuber observed critically that ROSA-IV will generate conservative as well as non-conservative results. I believe test results must be understood, but the aspect of conservatism is irrelevant for gathering test data for code assessment.

The work we have performed so far is aimed at evaluation of ROSA-IV as a facility that should provide data to assess capabilities of computer codes to model the overall system behavior during postulated transients. Our analyses showed that ROSA will exhibit most of the AP600 processes. The sequence and relative magnitude of events will be reproduced by ROSA-IV for most transients. However, ROSA will not be a AP600 simulator or a demonstration facility. Based upon our analyses, we believe that ROSA-IV can provide very useful data for assessment of computer codes to simulate AP600 integral system behavior during high pressure and depressurization phases.

On page 11 of his memorandum, Dr. Zuber discusses the issue of asymmetrical behavior of AP600 and respective code capabilities: "...RELAP5, being a one-dimensional code, cannot properly model the effects of flow asymmetries." Because the ROSA shortcomings in simulation of the asymmetries was considered as the most important, I would like to discuss this aspect in more detail.

Potential asymmetry in AP600 response is associated with flows in such components as the cold legs, pressure balancing lines, direct vessel injection (DVI) lines and downcomer. The asymmetry is a result of interactions between these components for a set of transients. Most of the DBA transient will be of a symmetrical nature in AP600 with respect to safety systems response. Only transients with breaks in DVI lines or pressure balancing lines will exhibit asymmetrical behavior not typical for current generation reactors.

Flows in all the components of interest, except in the downcomer, can be treated one-dimensionally as it is practiced for present generation reactors (all current system codes such as TRAC or RELAP treat piping flow as one-dimensional). Because of potential multi-dimensional effects in the downcomer we have nodalized it applying interconnected mesh of cells. However, to increase the fidelity of simulation, we suggested a rigorous two-dimensional modeling of the downcomer. Furthermore, we are currently analyzing the downcomer behavior of both systems with a 3-D CFD code (FIDAP), to obtain an independent evaluation of ROSA/AP600 comparison.

Current analyses indicate that ROSA-IV can exhibit the local phenomena that control AP600 response. With the currently proposed configuration ROSA-IV will be able to provide data on symmetrical system response and on asymmetrical response. The asymmetrical transients will not duplicate AP600 system behavior but will provide, in an integral system environment, interactions and phenomena that govern the asymmetric AP600 behavior. Codes validated using these data should be able to simulate AP600 symmetric and asymmetric system response.

Sincerely,

S. M. Modro
NRC Thermal Hydraulic
Analysis Programs

dap

cc: D. Bessette, US/NRC
W. H. Rettig, DOE Field Office, Idaho, MS 1134
J. C. Okeson, EG&G Idaho, MS 3600

APPENDIX II

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Nuclear Safety Research Review Committee
Washington, D.C. 20555

[illegible]2 July 1992

Mr. Eric S. Beckjord
Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Beckjord:

Enclosed please find a copy of a letter report of NSRRC's ALWR Subcommittee on AP600 thermal hydraulic testing. This letter report has been received and reviewed by the NSRRC and is accepted as a statement of the Committee's current position on AP600 thermal hydraulic testing.

If you have any questions on this NSRRC report, please contact Dr. Neil Todreas or me.

Sincerely,

David L. Morrison
David L. Morrison
Chairman
Nuclear Safety Research Review Committee

DLM/sje

Attachment

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
Nuclear Safety Research Review Committee
Washington, D.C. 20555

July 20, 1992

Dr. David Morrison
The MITRE Corporation
7525 Colshire Drive, MC W766
McLean, VA 22102

Dear Dr. Morrison,

The NSRRC Advanced Reactors Subcommittee (Messrs T Boulette, S. Burstein, H. Isbin and N. Todreas (Chairman) in attendance) met on July 1 and 2, 1992, and reviewed Office of Nuclear Regulatory Research (RES) programs pertaining principally to the AP-600 program. Among these programs, the RES proposal to conduct integral systems tests at the ROSA facility of the Japanese Atomic Energy Research Institute (JAERI) was examined in detail. Because of the timeliness of this NSRRC review regarding the forthcoming Commission decision whether to proceed with this program, this letter has been prepared to set forth the relevant conclusions of our review. A supplementary report of the full scope of the July 1 and 2 meeting will follow which will contain detailed observations and suggestions relevant to the RES programs examined.

THE RES PROPOSAL

The RES proposal examined was to conduct USNRC sponsored confirmatory integral systems tests on AP-600 using a full-pressure, full-height facility. The purpose of these tests is to develop a sufficient data base with which to enhance the assessment of an analytical tool that could then be used with confidence to assess full size plant responses to initiating accident sequences. The selection of a facility is constrained by the: (a) desire to obtain test results prior to the currently scheduled preparation (Summer, 1994) of the Draft Safety Evaluation Report (DSER) and the issuance (November, 1994) of the Final Design Assessment (FDA), and (b) need to obtain these results within the currently anticipated budget for this work of approximately $10 million. The selected facility is ROSA modified as proposed by RES and agreed to by JAERI to a configuration (ROSA V) representing a 1/30 by volume scaled model of AP-600 with the major model deviation being the use of a single versus the actual two cold legs per loop.

The NSRRC Subcommittee examined this proposal by posing and resolving a series of questions, starting with the need for this testing and culminating in the examination of the efficacy and adequacy of the proposed solution. These questions, restated specifically for AP-600 integral systems testing, will be sequentially reviewed next by summarizing the NRC position and then stating the Subcommittee's conclusion.

"What are the NRC's needs for confirmatory systems research on AP-600"?

Integral systems tests in a full-pressure, full-height facility are deemed necessary because the response of the systems to initiating events cannot be analytically predicted with confidence by the use of existing analytical tools (computer codes). This is due to both the possibility of interactions between systems and components and the low driving heads uniquely inherent in the passive system design as currently proposed. The test data are to be

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used to qualify an analytical tool to assess plant response. This approach is taken because no scaled facility can serve as a demonstration of full size plant response to initiating events. System behavior under three accident sequences is of particular interest because the passive safety systems are called upon to operate at high pressure:

- Small break loss-of-coolant accident,
- Steam generator tube rupture, and
- Steam line break.

Independent NRC testing at low pressure is not considered essential since the planned vendor test program is deemed to yield sufficient data. However, it is anticipated that the high pressure ROSA facility to be used by the NRC as discussed later can be run to yield supplementary lower pressure data.

The NSRRC Subcommittee concurs with the NRC's need for independent confirmatory systems research to insure that its analytic tools are qualified to assess full plant response. The availability of integral systems low pressure data to insure performance of the gravity drain/core cooling system behavior is recognized as equally important as high pressure test results.

"What integral systems testing program has the vendor proposed for AP-600"?

The high pressure test program will be conducted in the full-pressure, full-height, 1/395 by volume, scaled SPES-2 facility in Italy. A low pressure (400 psi maximum) test program will be conducted in the 1/200 by volume scaled Oregon State University (OSU) facility. The extension of SPES tests below 400 psi so as to initialize OSU tests is being explored.

The NSRRC Subcommittee took note of this planned vendor test program.

"Why should the NRC conduct confirmatory integral systems tests on AP-600 using a full-pressure, full-height facility when the vendor will conduct a similar test program"?

The NRC stated that they had a need to extend the expected vendor test matrix beyond the design basis to develop confidence that the design basis is a satisfactory limit. This would be achieved by experiments at or slightly beyond design basis conditions to ensure that no unanticipated phenomena or major effects occurred in this operating band, and thereby confirm the adequacy of the design basis limit.

The NSRRC Subcommittee concurs with the NRC need to develop confidence in the design basis in this manner. However, it is emphasized that we do believe that vendor demonstration of satisfactory plant performance within the design basis should clearly remain the required standard for design approval.

"Why did the NRC select ROSA as the test facility rather than use the Italian SPES facility in which the vendor will conduct tests or construct a new, domestic facility"?

The NRC could have chosen to contract separately with the SPES operators for conduct of an independently prescribed NRC test matrix; thereby avoiding conflict of interests. This was not done primarily for two reasons:

- Vendor access to the facility will take precedence over NRC access. Delay in conducting the vendor program or extension of the vendor program is possible and would severely upset the NRC schedule for acquisition of NRC independently produced test data. The value of the test results will be maximized if they can be used in the assessment of codes required for NRC safety analyses.

- The vendor has not presented analysis to the NRC to firmly establish that the data from SPES is valid by itself to qualify an analytic tool for use on a full scale plant. Scale effects probably need to be assessed and confirmed, as they have in past NRC thermal/hydraulic test programs, by tests at different scales.

The NSRRC Subcommittee concurs that plans to conduct NRC tests in SPES would not be prudent because of the cited schedule and test scale concerns.

"What is the NRC doing to ensure that the ROSA facility will be configured correctly and will simulate the performance of the AP-600 passive safety features with acceptable fidelity"?

The NRC has performed an extensive comparative assessment, using the RELAP 5, MOD 2.5 analytic tool, of the behavior of the ROSA facility and the AP-600 plant to the same set of initiating events. From these analyses, desired improvements in the ability of ROSA to simulate the phenomena appearing in the plant were identified. Costs for these improvements, specifically changes in the facility configuration, were estimated and subsequently negotiated with the ROSA owner. The final negotiated configuration has been analyzed and is expected to satisfactorily represent all full plant phenomena including many, but not all, aspects of asymmetrical loop behavior. The cost for ROSA modifications and the schedule for their implementation and the conduct of the test program meet NRC criteria. Further, the NRC stated to us that no domestic facility could come close to meeting the NRC cost and schedule criteria in that a cost of $40-50 million and a time of approximately three (3) years would be required to construct a domestic facility meeting or improving on the ROSA V facility criteria.

The NSRRC Subcommittee reviewed the technical basis for the proposed modifications to ROSA and its consequent suitability as the NRC's selected high pressure test facility. The Subcommittee concludes that the following key factors need to be balanced in reaching a decision:

- The importance of obtaining independent NRC data to confirm the adequacy of the design basis limit.
- The advantage of obtaining these data in a timely manner to allow their use in assessing codes used in safety analyses.
- The need to avoid the possibility of introducing ambiguity into the assessment process from experimental data taken on a test facility which may not represent full plant phenomena in all aspects.

After reviewing the data presented[1] and weighing these factors, the Subcommittee concurs with the RES recommendation to proceed with the ROSA V program for integral systems testing of the AP-600 plant design. This activity needs to be part of a well integrated

[1] During the preparation of this report, the Subcommittee received and reviewed the comments of the ACRS consultants concerning the SPES and ROSA integral test facilities.

program involving careful code enhancement and assessment, and possibly well-selected separate-effects tests for phenomena that cannot be fully explored in these integral facilities. Such a program is needed since the purpose of the integral testing is not a demonstration of AP-600 performance, but rather it is to gather data for code assessment. These aspects will be discussed more fully in our supplemental report.

Sincerely,

Neil Todreas

Neil E. Todreas
Chairman, NSRRC
Advanced Reactors Subcommittee

APPENDIX III

Appendix III

Subject: Reply to Memorandum by V. K. Dhir on the June 3-4 Meeting in Idaho Falls

The following responds point-by-point to comments provided by Dr. Dhir on the subject meeting. It is organized according to Dr. Dhir's memo.

Meeting Summary

No comment.

Observations

1. The ROSA tests are planned to cover the full pressure range from full system pressure to IRWST injection. They will, thus, overlap both the SPES high pressure and the OSU low pressure testing. We have stated we are interested in both high pressure and low pressure confirmatory testing. We explained to the ACRS that at the time when the staff was proposing to do separate low pressure confirmatory testing, the facility we envisaged was identical to OSU. At the time we were precluded from interacting with OSU due to conflict of interest considerations. Once these were resolved, we could no longer justify pursuing a duplicate facility.

   In contrast, based on our interest in high pressure systems interactions phenomena and processes, by June, 1991 we identified ROSA as the best candidate for performing high pressure confirmatory testing. We have been working since that time towards formulating a technically sound program to modify the facility and conduct testing.

   In terms of experimental programs in scaled facilities intended to model full scale power reactors, it is a well-established principal that facilities of different scales and scaling approaches should be used to ensure that the effects of scale are well-understood. By definition, scaling introduces distortions in all scaled facilities. Testing programs must be formulated accordingly. The integral system test program carried out to study small break LOCAs in Babcock and Wilcox reactors was an example of such a program. Research Information Letter 164, describing the results of this program, is attached. We would like to refer Dr. Dhir to Commissioner Rogers' memorandum (attached) on this program. In addition, to quote from MIT Professor Peter Griffith's review of the ROSA program:

   > "This is a well structured program which clearly benefitted from our experience with the LOCA work done on LWRs during the 70's and 80's. Because the experimental program consists of three integral tests being run on three quite different rigs

each scaled according to three different rationales, I can't imagine that there will be many questions outstanding about the system performance when these experiments are completed.

We have, in addition, got an operating, documented computer code, RELAP-5, which can be used for the prediction and analysis of the ROSA-IV experiments. The pieces of the program will come together in a timely way so that the results of this program can be used to design out any problems which might arise in the course of this research."

2. The costs and schedule would be effectively prohibitive. We have already given serious consideration to a domestic facility. We have stated that, ideally, this is our preference, however, such a facility could not be built within the FDA schedule, and would be considerably more expensive than the ROSA program.

   In its letter of March 10, 1992, the ACRS stated "Inasmuch as FHFP integral system testing will require at least three to four years to complete, there is a risk that the present certification schedule will be affected unless the test program is begun now. We believe the likelihood of such an impact is great. If the present certification schedule is to be adhered to, we recommend that a FHFP testing program be initiated now." RES agrees with the ACRS' assessment.

3. The use of RELAP to perform comparative calculations of ROSA and AP600 is not circular and certainly is logical. We completed a RELAP code applicability review in early 1991. RELAP models and correlations were reviewed from the perspective of new AP600 phenomena and features that could be important to reactor safety. The purpose was to identify those areas in which new mathematical models of physical phenomena would be required to be added to RELAP5. In most cases, the AP600 design and its systems and the planned and off-normal operations were found to be similar enough to current PWRs that RELAP safety analysis applicability was unchanged. There were basically no new phenomena involved in the AP600, however the physical parameter ranges and applications of the phenomena may be different than those in present generation reactors. Therefore, a validation program was laid out accordingly.

   Review of ROSA with respect to our separate scaling study for a FHFP facility showed that ROSA meets all requirements. If Dr. Dhir is aware of a better method to demonstrate facility similitude and scaling adequacy, we should like to know of it.

4. Distortions will be present in ROSA, as they will be in each and every scaled facility. The true question is whether the major phenomena and processes are preserved. Our analyses have shown that they are.

   In addition, JAERI has agreed that the facility will remain available beyond the initial set of 10 experiments.

5. The experiments must meet the objectives and specifications determined prior to the experiments. Otherwise, they will be repeated.

6. RES has assessed the cost-benefit ratio for the ROSA program, along with schedule requirements of the FDA, and found that this facility meets all requirements identified by NRC staff and contractors for confirmatory integral system testing.

7. We never planned to stop the experiments at the actuation of 3rd stage of ADS. Rather, we plan to run the ROSA experiments to full depressurization, including the initiation of IRWST injection.

8. Aside from the problem of schedule, we could not achieve the same data base for the same cost with a new U.S. facility.

9. In an ideal world, we would also prefer a dedicated U.S. facility, however, this is not possible within the constraints of schedule and budget. Pragmatic alternatives must be sought that will successfully meet the same needs within the given constraints.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D C 20555

JAN 1990

MEMORANDUM FOR: Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

FROM: Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

SUBJECT: RESEARCH INFORMATION LETTER 164, "THERMAL-HYDRAULIC DATA BASE RELEVANT TO PLANTS OF THE BABCOCK AND WILCOX LOWERED-LOOP DESIGN"

References:
1. ALAB-708, 16 NRC1770, December 29, 1982.
2. Clarification of TMI Action Plan Requirements, NUREG-0737, November 1980.
3. Letter from H. R. Denton to R. B. Minogue, "Request for the Conceptual Design of a Facility for the Study of B&W and CE Integral System Characteristics," December 30, 1981.
4. J. Gloudemans and D. P. Birmingham, "MIST Program: Summary of Key Results," NUREG/CP-0097, Vol. 4, March 1989.
5. K. Almenas, et al., "Scaling of Integral Facilities at Reduced Pressures," MDNE/061589, June 15, 1989.
6. K. Almenas, et al., "Evaluation of Four MIST Atypicalities," MDNE/041089, April 10, 1989.
7. Letter from H. R. Denton to R. B. Minogue, "Request for Follow-on Program in the B&W Integral System Test Facility (MIST)," October 31, 1984.

This memorandum transmits results from research conducted in the Integral System Test (IST) program. IST includes the Multi-loop Integral System Test (MIST) facility at the Alliance Research Center in Alliance, Ohio, and the University of Maryland at College Park (UMCP) 2x4 Loop facility. This research provided thermal-hydraulic experimental data relevant to plants of the Babcock and Wilcox (B&W) lowered-loop design. MIST was jointly sponsored by the U.S. Nuclear Regulatory Commission, the Electric Power Research Institute (EPRI), B&W Owners Group (B&WOG) and B&W. The UMCP 2x4 Loop, a reduced-pressure and small-scale facility, was designed to address scaling atypicalities of the MIST facility and to provide data for code assessment.

Contacts:
R. Y. Lee, RES/DSR, 49-23560
H. Scott, RES/DSR, 49-23563

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Regulatory Issue:

Following the Three Mile Island Unit 2 (TMI-2) accident, a number of regulatory issues concerning the design of the B&W reactors were raised. The effectiveness of feed and bleed and the boiler condenser mode (BCM) of natural circulation was challenged during the TMI-1 Restart Hearing conducted by the Atomic Safety and Licensing Appeal Board [1]. In BCM, heat is removed from the primary system through vapor condensation in the steam generator and the accompanying primary-to-secondary heat transfer. Clarification of TMI Action Plan Requirements (NUREG-0737) Item II.K.3.30 [2] required that small-break loss-of-coolant accident (LOCA) calculational models be compared to applicable data.

In response to NRR's request for integral system characteristics for B&W reactors [3], the NRC and industry formed a Test Advisory Group to make recommendations regarding the type of data base required to validate small-break LOCA models. The IST program was formed in 1983 to acquire the desired data. The primary experimental facility in the IST program is the MIST facility.

Although MIST is designed as a full-height and full-pressure integral experiment facility, it is still a scaled model of a B&W plant. Thus, it entails various design compromises such as an atypical downcomer. These design compromises are a potential source of distortion of some of the physical phenomena (e.g., variation of flow regimes in the hot legs) which in turn could lead to atypical transient behavior (e.g., premature interruption of natural circulation). The UMCP 2x4 Loop, a reduced-height and reduced-pressure, integral experiment facility employed an alternate design approach (e.g., a more typical downcomer) to assess the impact of some of the MIST design compromises on transient behavior.

Conclusion:

IST produced an integral experiment data base for natural circulation, small-break LOCA, feed and bleed, steam generator tube rupture, effects of non-condensible gases, and pump operations on small-break LOCA behavior [4, 5]. Key observations (1 to 6 for MIST, 1 and 7 for UMCP) are summarized below.

(1) **Natural circulation was studied under varying degrees of loss of primary inventory. A key question about natural circulation was whether the BCM would remove decay heat effectively and depressurize the reactor coolant system. This mode of heat transfer was consistently observed.**

(2) **During small-break LOCA, heat removal from the primary system was further augmented by the steam venting from the upper plenum to the downcomer through the reactor vessel vent valves (RVVV). Adequate heat removal was observed in MIST tests for a wide range of primary boundary conditions (i.e., variation of break sizes from 5 to 50 cm^2, variation of break locations, and both full- and half-capacity HPI flows). In all cases tested, the MIST system depressurized and attained primary circuit mass equilibrium without uncovering the core.**

(3) **MIST results show that the feed and bleed technique can be utilized to cool the core and depressurize the primary system.**

(4) For multiple simulated steam generator tube ruptures, the primary system depressurized rapidly due to tube-rupture discharge. BCM-like activity was observed between primary system break flow to the secondary side of the ruptured steam generator. For a smaller number of tube ruptures, the primary system depressurized by single-loop cooldown of the intact loop.

(5) The presence of non-condensible gases reduced BCM cooling, but did not prevent primary system cooldown and depressurization.

(6) Reactor coolant pump operation is advantageous during small-break LOCA. With forced flow, primary-to-secondary heat transfer is maintained longer and more energy is removed from the break. Therefore, the primary system pressure decreases more rapidly and the primary system refills more quickly.

(7) The comparisons of the experimental results from the two facilities indicated that the UMCP 2x4 Loop is able to simulate the thermal-hydraulic behavior observed in MIST. First, it reproduced the qualitative aspects of the flow modes. That is, it exhibited similar local flow regimes, flow regime transitions, the presence of both steady state and boiler-condenser natural circulation flows, and loop asymmetries [5]. Second, it reproduced the sequence at which these flow modes occur during an inventory depletion transient. Inventory scaling was used to estimate the quantitative aspects of the flow modes (e.g., duration, magnitude of pressure changes). A precise parameter to parameter mapping between the UMCP and MIST data is not implied and is, in fact, precluded by the stochastic nature of some flow mode transitions. However, key phenomena of inventory transients can be simulated and the effect of pressure on the characteristics of these phenomena is understood. Despite the differences between the design of the two facilities, similar thermal-hydraulic characteristics were observed. The MIST design atypicalities do not affect the expected thermal-hydraulic behavior during a small-break LOCA [6].

Regulatory Implications:

The MIST and the UMCP 2x4 Loop experimental data provide a sufficient small-break LOCA data base to satisfy the requirements of NUREG-0737. The integral system data is self-consistent, comprehensive, and suitable for benchmarking computer codes used to calculate B&W plant transients. Such benchmark calculations for MIST were performed and are in good agreement with experimental data. The data, as well as code calculations, show that various methods, such as feed and bleed and BCM, are effective modes of decay heat removal in a B&W plant during a small-break LOCA.

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Restrictions on Applications:

The scaling evaluation has shown that key thermal-hydraulic behavior (e.g., BCM) observed in these facilities can be expected to occur in full scale B&W plants of the lowered-loop design. However, these test facilities are scaled models of a B&W lowered-loop nuclear steam supply system. As such, various scaling atypicalities in simulating a plant were required. Hence, these data should not be applied directly to a full scale plant. Rather, validated computer codes (TRAC-PWR, RELAP5) should be used to calculate plant small-break LOCA. The requisite code validation was performed as part of the IST program.

Further work:

At the request of NRR [7], additional testing was performed in the MIST facility to obtain data for: small-break LOCA without high pressure injection; station blackout; and examining scaling questions. Analysis of these tests is expected to be completed by the end of 1989. Additional experiments are being performed at UMCP to further test the scaling concepts under more complicated boundary conditions, i.e., with HPI flow. Any new significant results will be reported in a future RIL.

Eric S. Beckjord

Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

Enclosures:
(1) MIST Program: Summary of Key Results
(2) Evaluation of Four MIST Atypicalities