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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 22 1981

MEMORANDUM FOR: J. P. Knight, Assistant Director for Components and Structures Engineering

FROM: T. M. Novak, Assistant Director for Operating Reactors

SUBJECT: ACCEPTABILITY OF THE MILLSTONE, UNIT NO. 2 ENCLOSURE BUILDING AND ITS FILTRATION SYSTEM

Late in the construction of Millstone Power Station, Unit No. 2 (M-2), the staff stipulated that a further reduction in the off-site dose following a LOCA was necessary. Northeast Nuclear Energy Company (NNECO) responded by proposing an enclosure building (EB), a limited leakage steel framed structure, completely surrounding the containment above grade. This solution was found acceptable as stated in our May 10, 1974 Safety Evaluation Report.

In the later part of 1977, NNECO discovered and notified the staff that some enclosure building filtration system (EBFS) lines were not designed and supported seismically. These problems were corrected prior to the start of Cycle 2 operation. However, in a follow-up letter of March 1, 1979 (Enclosure 1), NNECO provided their conclusions based on additional investigation into the seismic design of the EB and the EBFS. They summarized that, "...the enclosure building structure is designed to retain structured integrity subsequent to a seismic event. However, the EBFS is not designed to be functional subsequent to an SSE". This statement means, according to conversations with NNECO, that the EB support frame and the entire EBFS will be intact following SSE, but the sheet steel may tear at some locations. Thus, maintaining a 0.25 inch negative pressure in the EB (the system design objective) is not assured after the SSE.

This problem was the subject of a meeting with you held in my office in late September 1980. At this meeting, we got the impression that the staff requirement, that seismic loads be combined with LOCA loads for a building performing a backup to containment function (such as the EB at Millstone-2), may be changing. This possible relaxation in the above stated requirement by your staff was also noted at the April 15, 1981 meeting on Asymmetric LOCA Loads, Unresolved Safety Issue A-2. We understand that justification for such relaxation stems, at least partially, from research reports such as NUREG/CR-1889 which documented the calculated probability of a combined LOCA and SSE at 1.8 X 10-12 (Enclosure 2 contains selected pages from this report).

Based on the above, it does not seem reasonable for the NRC to require NNECO to modify the EB to meet the combined LOCA plus SSE load combination criteria. If you agree with the preceding statement, TAC No. 11522 will be closed-out as completed and Mr. N. Romney of your staff should not charge any further review time to this TAC.

If, however, you disagree with our close-out of this issue on the Millstone-2 docket, please provide us the technical basis and schedule for completing your review of this issue. Your response within the next 15 days would be appreciated.

Thomas M. Novak, Assistant Director for
Operating Reactors
Division of Licensing

Enclosures: As stated

Enclosure 1

NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
THE HARTFORD ELECTRIC LIGHT COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

P.O. BOX 270
HARTFORD, CONNECTICUT 06101
(203) 666-6911

March 1, 1979

Docket No. 50-336

Director of Nuclear Reactor Regulation
Attn: Mr. R. Reid, Chief
Operating Reactors Branch #4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References: (1) D. C. Switzer letter to G. Lear dated September 22, 1977.
(2) G. Lear letter to D. C. Switzer dated October 12, 1977.
(3) D. C. Switzer letter to R. Reid dated March 13, 1978.

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Enclosure Building Design

In Reference (1), Northeast Nuclear Energy Company (NNECO) advised the NRC Staff that the ten-inch suction line to Fan F-55 was not designed and supported seismically. This situation was reported as a 24-hour Reportable Occurrence, on the basis of a postulated coincident safe shutdown earthquake (SSE) and design basis accident (DBA). Under such a circumstance, failure of this line could prevent the enclosure building filtration system (EBFS) from maintaining a 0.25 inch negative pressure in the EBF region. In Reference (2), the NRC Staff concurred with the assessment and the proposed corrective action, that of qualification of the line as seismic Category I prior to Cycle 2 operation. In Reference (3), NNECO indicated that two additional non-seismically supported lines in the EBF region were discovered. All modifications were completed prior to the start of Cycle 2 operation, as reported.

The purpose of this letter is to advise the Staff of recent investigations in this area, and report NNECO's conclusions.

NNECO has determined, based on further review, that the existence of non-seismically designed and supported lines which penetrate the enclosure building is acceptable. This situation is, in fact, the original design basis as reported in the FSAR and approved by the Staff. The response to FSAR Questions 6.9 and 6.16.4, provided in Amendments 39 and 16, respectively, discusses various non-seismic Category 1 penetrations through the EBF region. This configuration was, and continues to be, an acceptable design basis. As stated previously, the basis for the Reportable Occurrence of Reference (1) was a postulated coincident SSE and DBA, or DBA followed by an SSE. Although such a postulate was clearly a safe method to evaluate the adequacy of the design of Millstone Unit No. 2, it was also excessively conservative. Having recently identified certain additional non-seismic penetrations, NNECO has further reviewed and evaluated the original design basis as well as current regulatory guidance in this area, and concluded that the current configuration is acceptable as presented below.

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A thorough review of the Millstone Unit No. 2 FSAR Sections 5.3.3.1.3, 5.3.3.1.4, 5.3.3.2, and 5.3.3.2.1 reveals that although the enclosure building is designed to mitigate the consequences of a DBA and to maintain its structural integrity during and after a seismic event, simultaneous occurrence of the two events was not the design basis for the enclosure building.

The Staff's Safety Evaluation Report, dated May 10, 1974, states in Section 3.9 that "an appropriate combination of loads likely to occur" was considered in the design of the enclosure building. The NRC Staff did not require consideration of a simultaneous loss-of-coolant accident and seismic event for the enclosure building when the design was approved.

As stated in General Design Criterion 2 of 10CFR50, Appendix A, only "appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena" need be considered. For example, it is clear that tornado events were not required to be combined with either loss-of-coolant events or seismic events due to the low likelihood of simultaneous occurrence.

Appendix 5.B of the FSAR states that seismic events and loss-of-coolant incident were considered together, but only for the purposes of assuring conservatism in specifying appropriate load combinations used in design equations of the containment structure. These two events were not combined for the purposes of conducting a consequence evaluation regarding performance of the EBFS.

This logical approach of not combining remote probability events to evaluate consequences is evident in a number of regulatory documents. For example, Regulatory Guide 1.117, "Tornado Design Classification" states that "it is not necessary to maintain the functional capability of all Seismic Category I structures, systems, and components because the probability of the joint occurrence of a low-probability event (loss-of-coolant accident with DBT or smaller tornado, or earthquake with DBT or small tornado) is sufficiently small".

NNECO fully realizes that SSE and LOCA loads have long been combined by means of factored loads for the containment structure and reactor coolant piping system analysis. As indicated by the NRC Director of the Division of Systems Safety, Dr. R. J. Mattson, at the June 2, 1978, ACRS meeting, the reason for this conservative load combination is not clear to the NRC Staff at the present time. The NRC is presently developing a methodology for establishing the appropriate margins in the containment structure and reactor coolant piping system design to replace the arbitrary combination of loads from two remote probability, independent events (NUREG-0484, "Methodology for Combining Dynamic Responses").

To summarize, the enclosure building structure is designed to retain structural integrity subsequent to a seismic event. However, the EBFS is not designed to be functional subsequent to an SSE.

The hypothetical situation of coincident LOCA and SSE is beyond the scope of the original Millstone Unit No. 2 licensing basis. Further, such a postulate has insufficient technical basis, and, therefore, cannot be considered credible.

NNECO is not docketing this letter to request Staff review, but rather to clarify the misleading and excessively conservative interpretations we presented in References (1) and (2). We trust the above information is sufficient to clarify our position.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

W. G. Counsil

W. G. Counsil
Vice President

ENCLOSURE 2

NUREG/CR-1889
UCID-18694
RM

Large LOCA-Earthquake Combination Probability Assessment— Load Combination Program Project 1 Summary Report

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NRC FIN A0133

ABSTRACT

This report summarizes work performed for the U.S. Nuclear Regulatory Commssion (NRC) by the Load Combination Program at the Lawrence Livermore National Laboratory to establish a technical basis for the NRC to use in reassessing its requirement that earthquake and large loss-of-coolant accident (LOCA) loads be combined in the design of nuclear power plants. A systematic probabilistic approach is used to treat the random nature of earthquake and transient loading to estimate the probability of large LOCAs that are directly and indirectly induced by earthquakes. A large LOCA is defined in this report as a double-ended guillotine break of the primary reactor coolant loop piping (the hot leg, cold leg, and crossover) of a pressurized water reactor (PWR). Unit 1 of the Zion Nuclear Power Plant, a four-loop PWR-1, is used for this study.

To estimate the probability of a large LOCA directly induced by earthquakes, only fatigue crack growth resulting from the combined effects of thermal, pressure, seismic, and other cyclic loads is considered. Fatigue crack growth is simulated with a deterministic fracture mechanics model that incorporates stochastic inputs of initial crack size distribution, material properties, stress histories, and leak detection probability. Results of the simulation indicate that the probability of a double-ended guillotine break, either with or without an earthquake, is very small (on the order of 10^{-12}). The probability of a leak was found to be several orders of magnitude greater than that of a complete pipe rupture. A limited investigation involving engineering judgment of a double-ended guillotine break indirectly induced by an earthquake is also reported.

INTRODUCTION

BACKGROUND

The Code of Federal Regulations requires that structures, systems, and components important to the safety of nuclear power plants in the United States be designed to withstand appropriate combinations of effects of natural phenomena and the effects of normal and accident conditions.[1] Historically, the U.S. Nuclear Regulatory Commission (NRC)--through Regulatory Guides, regulations, branch technical positions, and the Standard Review Plan--has required that the responses to various accident loads and loads caused by natural phenomena be combined in the analysis of safety-related structures, systems, and components.

Designing safety-related structures, systems, and components to withstand the combined effects of an earthquake and a large loss-of-coolant accident (LOCA) is one such load combination requirement that has been implemented by the nuclear industry for more than ten years in the design of commercial nuclear power plants. The combination of the most severe LOCA load with safe shutdown earthquake (SSE) loads was not controversial until about five years ago when the postulated LOCA and SSE loads were both increased by a factor of two or more to account for such phenomena as asymmetric blowdown in PWRs and techniques to better define the loading were developed.

As a result of this change, the combination requirement became more difficult to implement, particularly in the design of reactor pressure vessel internal and support systems. For future plants, the change brought with it the prospect of increased construction costs. Additionally, the load combination requirement raises the issue of whether design for extreme loads will result in reduced reliability during normal plant operation. For example, present seismic design methods tend to result in stiff systems and more supports when additional strength is provided for the earthquake loading. Because a stiff system is subjected to greater cyclic thermal stress than a flexible one under normal thermal operating loads, reliability is reduced under normal conditions.

Faced with these design, cost, and safety issues, the nuclear industry petitioned the NRC to reconsider its design requirement. From a safety viewpoint, costs should not be a factor in changing design requirements. The costs of meeting design requirements are industry's responsibility. However, for existing plants to meet the revised loading definition and also satisfy the combination requirement, modification is almost unavoidable. Certain plants can be feasibly modified, but other plants are not feasible to modify, and they present a difficult problem to the NRC. The solution can be either to allow continued operation without modifications, challenge the safety of continued operation without modifications, or solicit a technical basis for eassessing the design requirement.

Industry's approach to the problem has been to assure safety by justifying that the combination of events is unlikely. For the NRC to accept industry's justification and change the design requirement, independent confirmatory research is necessary. One such study has already been made for the NRC by Battelle Columbus Laboratories.[2] The Battelle researchers used a deterministic approach to assess the likelihood of a break occurring in a cold leg pipe of a PWR plant. The work reported herein assesses the hot leg, cold leg, and crossover of a PWR plant. The approach used is probabilistic. Both efforts will be considered by the NRC in assessing the nuclear power industry's submittal and request.

OBJECTIVE

The objective of this study is to estimate the probability that a large LOCA and an earthquake occur simultaneously. This information will be used by the NRC to reassess the requirement that earthquake and LOCA loads be combined. If a LOCA and an earthquake are independent events, the probability of simultaneous occurrence is expected to be very low. However, if an earthquake can cause a LOCA, the probability of simultaneous occurrence may be significant. Thus, this assessment considers only LOCAs that are directly and indirectly induced by earthquakes in addition to normal and abnormal plant operating loads.

SCOPE

In Phase I, we limit our investigation to determining the probability of a large LOCA induced by earthquakes for a pressurized water reactor (PWR) plant. A large LOCA is defined as a double-ended guillotine break of the primary reactor coolant loop piping--the hot leg, cold leg, and crossover. Such pipes typically have outside diameters of 30 inches or more, and have walls that are approximately 2.5 inches thick. This evaluation is limited to the rupture of such large pipes because they will generate the most severe LOCA loads, which, when combined with SSE loads, present the design and retrofit problems discussed above.

We recognize that the break of a smaller pipe may be more probable, and that such a small LOCA may pose larger risks to the plant. However, for Phase I we limit our scope to the large LOCA defined above in order to address the immediate NRC need for confirmatory research. We believe that the models and computational procedures developed for the large LOCA can be extended to the assessment of smaller LOCAs during subsequent phases of our study.

Only fatigue crack growth resulting from the combined effects of thermal, pressure, seismic, and other cyclic loads was considered as the mechanism leading to complete pipe rupture as a direct consequence of earthquakes. The water hammer mechanism was not considered because it has never been observed in PWR primary systems. Likewise, stress corrosion is another plausible mechanism, but it was excluded from consideration because the coolant water chemistry in PWRs is such that stress corrosion problems have not been observed.

A limited investigation involving engineering judgment of a double-ended guillotine break indirectly induced by an earthquake is also reported. Earthquake-induced indirect causes such as falling cranes, mechanical, electrical and structural failure, as well as fire, explosion, and missiles are considered. The emphasis of this work was to identify sources and establish the ground work for a more thorough evaluation in a subsequent phase. Therefore, the preliminary results presented in this report are limited.

Unit 1 of the Zion Nuclear Power Plant (Zion Unit 1) was selected as the demonstration plant for this study. The results and conclusion are applicable only to Zion Unit 1 at this time; that is, no attempt has been made to extend our findings to other plants during Phase I. However, the methodology developed for the evaluation is an advanced computational tool. It can be applied in future evaluations to the break of reactor coolant pipes, large or small, to other PWR and boiling water reactor (BWR) plants, or to general piping reliability assessments. Such studies are clearly beyond the scope of the work reported here, but they may be part of future phases of the program.

APPROACH

The current practice of considering these dynamic events acting concurrently has generally been based on conservative engineering judgment that has not addressed the fact that the postulated LOCA and earthquake loads are random events. Amplitude, duration, frequency content, time of occurence, and time-phase relationship are random and stochastic in nature. Thus, a systematic probabilistic assessment is necessary before a technical basis for an appropriate combination requirement can be developed.

A multiphase, systematic probabilistic approach was used to treat the random nature of earthquakes and LOCAs. In reaching our Phase I objective, we took the following steps:

- Considered many mechanisms that can lead to a pipe failure as a direct result of an earthquake, but concluded that fatigue crack growth resulting from the combined effects of thermal, pressure, seismic, and other cyclic loads has the highest potential to lead to complete pipe rupture. In particular, the water hammer effect was not considered because it has never been observed in PWR primary systems.

- Modeled fatigue crack growth with a deterministic fracture mechanics model that incorporated stochastic inputs of initial crack size distribution, material properties, stress histories, and leak detection probability.

- Developed structural models to calculate seismic stresses and nonseismic stresses induced in the piping by dead weight, pressure, thermal expansion, and transients. Steady-state vibrational stresses and residual stresses were also considered but found to be insignificant for growing fatigue cracks.

- Calculated the probabilities of pipe leaks and breaks during the plant's life by inputing to the fracture mechanics model the results of the stress analysis and estimates of the crack size distribution, material properties, and crack and leak detection probabilities.

- Estimated the probability of a directly induced large LOCA based on the site-specific seismic hazard.

- Performed limited sensitivity studies on the inputs to the stress analyses, fracture mechanics model, and estimation procedure.

- Used engineering judgment to estimate the probability of a large LOCA induced indirectly by earthquakes. Structural, mechanical, and electrical failures, as well as explosions, fires, and missile incidents caused by earthquakes were considered. The emphasis of this work was to identify sources and establish the ground work for a more thorough evaluation in a subsequent phase. Therefore, the preliminary results presented in this report are limited.

MAJOR ASSUMPTIONS

Assumptions are necessary to simplify the complex assessment. These assumptions and simplifications reflect engineering judgment that is based on information available from the literature and from preliminary analyses. The assumptions for the directly induced LOCA problem are listed below, with the most basic assumption listed first.

- Failure results from fatigue crack growth of crack-like defects that are confined to the girth-welded butt joints. Note that these cracks are modeled in the circumferential orientation because of its reference to the postulated double-ended pipe break. Longitudinal cracks, though important to the leak assessment, will not result in such a guillotine break and are, therefore, not considered.

- Crack size distribution can be adequately characterized, despite the lack of data on large cracks. The tail of the crack size distribution (i.e., the large cracks that are present from the onset of crack growth) plays a major role in determining the probability of a LOCA. Unfortunately, there are no data on large cracks; thus, crack distribution data generated from smaller crack sizes must be extrapolated into this regime. Several distributions were considered,[3,4,5] and the most conservative distribution (that of Marshall, Ref. 3) was used.

- Based on observed pipe cracks, the crack shape is semielliptical; the shape is maintained during crack growth; and the length-to-depth aspect ratio is two or more. Only surface cracks are evaluated. Based on equivalent crack surface area and crack length, a surface semielliptical flaw has a stress intensity 1.4 to 2.5 times that of the embedded flaw. When this is cast in terms of fatigue behavior, we find that the surface flaw will grow at a rate of 4 to 50 times that of the subsurface flaw.

- No distinction is made between the shop or field welds for either material properties or crack size distribution. Data on fatigue and tensile properties of welded and unwelded 316 stainless steel show little variation in mechanical behavior.[6,7,8,9]

- The initial crack size distribution is independent of location within a joint and from joint to joint.

- At most, one crack is modeled in each joint, and crack interaction is not considered. Multiple cracks are ignored because the probability of having exactly one crack in a joint is approximately 0.09, whereas the probability of having two or more cracks in the same joint is less than 5×10^{-3}. These approximations are based on the probability of having a crack in a unit volume equal to $10^{-4}/\mathrm{in.}^3$ (an adaptation from Refs. 10 and 11) and a weld volume of approximately 10^3 in.^{3} per weld joint.

- Subcritical crack growth results from fatigue. Stress corrosion cracking is not considered because it has not been observed in PWR primary coolant piping.

- A Paris model describes crack growth.[6,12,13,14] Crack retardation stemming from pulses of high-stress cycles is not considered, nor is the enhanced fatigue behavior resulting from large values of stress intensity. The effects of material variation, weld properties, and environment are accounted for in the distribution of the coefficient to the Paris equation. When an inside surface crack grows through the pipe wall, its length on the outside surface is conservatively set equal to the inside length. Therefore, the minimum through-wall crack is at least twice the pipe thickness.

- Only cracks in the circumferential orientation are considered. Probable crack locations that are not considered in this study include longitudinal welds in elbows and pipe sections, nozzle corners, bimetallic transition joints (i.e., the reactor pressure vessel to safe-end welds), and base metal. Except for the bimetallic transition joint, these crack locations will not result in the double-ended pipe break. Data for fatigue of the bimetallic joint are unavailable. However, if the 316 stainless steel fatigue relation is a reasonable approximation of the bimetallic property, then the system failure probabilities should not change significantly.

- Stresses used in this evaluation include pressure, thermal expansion, dead weight, operating and abnormal transients, and seismic events. Conservative values of design temperature and pressure changes and frequency are used in calculating stresses. Vibration stress magnitudes are very low, and for crack sizes of interest result in a stress intensity below the threshold value. Residual stresses are found to be compressive on or very near the inside surface, and the stress intensities evaluated from them are negative. Residual stresses are not included because (1) the negative stress intensity would retard crack growth, and (2) there is a certain amount of uncertainty associated with residual stresses and how they redistribute as a function of crack growth.

- Design, construction, and assembly errors are not estimated. The primary piping is subjected to high quality assurance testing and if gross design or construction errors exist, they would be found during system checkout and hydrotesting.

- Support stiffnesses and structural damping values are estimated based on the best available data. All component supports and snubbers are assumed to be in good working order. The soil model and seismic hazard curve are both based on site-specific estimates.[15,16]

- Transients occur as a stationary Poisson process reflecting the Zion Unit 1 operating history for its seven years of operation. Earthquakes characterized by a peak horizontal ground acceleration less than 0.07 g are assumed to affect crack growth negligibly. The low stresses calculated bear this out. The maximum free-field peak acceleration that can be generated at the Zion site from the available data base is 0.85 g (5 x SSE).

- Both leak and large LOCA failures are considered. Cracks which grow through the piping wall lead to leaks of greater than 2 gpm based on the minimum crack length of twice the wall thickness. These leaks are assumed to be detected and repaired. The crack size distribution is set equal to the initial size distribution after weld repair.

- A large LOCA can occur if the load-controlled stresses are high enough to sever the pipe. It is assumed that dead weight, pressure, and seismic stresses are load controlled. Leaks that occur and lead to a LOCA during a seismic event are not evaluated--only the final result (leak or LOCA) after the earthquake is considered

- Pipe severance results from net section plastic instability, which occurs when the net section stress exceeds the average flow stress; $\sigma_{flow} = (\sigma_y + \sigma_u)/2$. The flow stress is assumed to be normally distributed. Approximation of the J-integral and the applied tearing yield considerably larger crack sizes.

- The simulation is ended when the first operating basis or larger earthquake occurs.

SUMMARY OF RESULTS

Results of the simulation indicate that the probability of a large LOCA, either with or without an earthquake, is very small. The probability of a large LOCA induced by plant transients given the condition that no earthquake occurs during the 40-year plant life is estimated to be 1.6×10^{-12}. Seismic stresses tend to increase this probability; that is, the probability of a large LOCA induced by plant transients and earthquakes is estimated to be 1.8×10^{-12}, assuming that the plant is shut down after an operating basis or larger earthquake. The probability of an earthquake and an earthquake-induced large LOCA occurring simultaneously during the plant's life is about 4×10^{-13}. These findings are consistent with other studies of this nature.

These results are best estimates based on the approach taken and the assumptions made. A limited sensitivity study on input variables was performed. Input variables were assigned incredible values to determine how the results vary for the following cases:

- No preservice inspection
- No preservice hydrostatic proof test
- No leak detection
- Modified Marshall initial crack distribution
- Variation of aspect ratio of the length and the depth of the initial crack
- Increase the 5 x SSE seismic stress by a factor of 3.

The limited sensitivity study indicates that leak detection capability, preservice inspection, and proof testing tend to reduce the probability of a LOCA by less than an order of magnitude. Increasing the percentage of cracks with large aspect ratio increases the large LOCA probability significantly; however, it has little influence on the leak probability.

Results of the simulation indicate that an earthquake has almost no effect on the leak probability, which is estimated to be 8×10^{-7} during the plant's life.

In view of the extremely low probabilities reported above, it can be concluded that a reactor coolant loop pipe break as a result of fatigue crack growth is highly unlikely. The probability that an earthquake and a LOCA induced by an earthquake occur simultaneously is less. These conclusions imply only that a reactor coolant loop pipe is unlikely to break as a direct result of the combined loads from normal plant operation and an earthquake. Other external earthquake-induced sources--such as fires, explosions, missiles, and structural, mechanical, or electrical failures--can potentially cause a pipe break as an indirect result of an earthquake. A limited investigation of such LOCAs indirectly induced by an earthquake revealed that most sources do not result in a LOCA which falls within our definition of a large LOCA. Several scenarios of possible sources were identified, and engineering judgment was used to estimate the probabilities of occurrence for the scenarios. A more refined assessment is needed if the indirect scenarios are to become part of the technical basis for decision-making.