ACTIVITIES

IN THE CZECH REPUBLIC

FOR

REACTOR PRESSURE COMPONENTS

LIFETIME MANAGEMENT

Milan BRUMOVSKÝ

NUCLEAR RESEARCH INSTITUTE ŘEŽ plc

250 68 ŘEŽ

to be presented at the meeting of the

IAEA International Working Group on

NUCLEAR POWER PLANT LIFETIME MANAGEMENT

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Vienna

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GENERALLY

The situation in NPPs after the splitting of the Czech and Slovak Federal Republic is :

Czech Republic	Slovak Republic
WWER-440/V-230	
	J.Bohunice - 2 units (operation)
WWER-440/V-213	
Dukovany - 4 units (operation)	J.Bohunice - 2 units (operation)
	Mochovce - 2 units (construction)
WWER-1000/V-320	
Temelín - 2 units (construction)	

Industrial, technical and research capacities were developing for a long time together in such a way to be complex in the whole Federation. Thus, it is not possible to cancel all common and mutual activities as well as cooperation.

Even though no more common state supported research and development programmes exist, cooperation in the field of reactor pressure components safety and lifetime management still exists in a wide form. The main reason can be found in the fact that Nuclear Research Institute as well as main components manufacturers (ŠKODA - RPV, VÍTKOVICE - SG and P, MODŘANSKÉ STROJÍRNY - PP) are located in the Czech Republic and most of activities started still before the federation splitting.

Main activites can be divided into the following fields :

- UPGRADING AND SAFETY ASSURANCE OF NPPs WITH REACTORS OF WWER-440/V-230

- SAFETY ASSURANCE OF NPPs WITH REACTORS OF WWER-440/V-213

- LIFETIME MANAGEMENT PROGRAMME OF NPPs WITH REACTORS OF WWER-440/V-213

- PREPARATION OF START-UP OF NPPs WITH REACTORS OF WWER-1000/V-320

- PREPARATION OF GUIDES FOR LIFETIME AS WELL AS DEFECT ALLOWABILITY EVALUATION IN MAIN COMPONENTS OF PRIMARY AND SECONDARY CIRCUITS

LIFETIME MANAGEMENT PROGRAMME FOR NNPs WITH WWER-440/V-213 REACTORS

MAIN TASKS :

- Procedure for residual lifetime assessment of key components of primary and secondary circuits for continous as well as for experts'level of evaluation
- Procedure (including software) for volume and methods for data collection and keeping that are necessary for continous as well as for experts' assesment
- Computation model of materials properties degradation of key components with respect to different ageing processes including recommendations for necessary diagnostics methods
- Analysis of the effect of individual as well as synergestic ageing processes and operating loading to residual lifetime of components
- Recommendation for operational regimes improvements, reconstruction of components with the aim to management of ageing and activ influen... of residual lifetime of components and a whole NPP
- Procedure for surveillance of operational reliability of key components of primary and secondary circuits for checking of a proposal for management of component ageing
- Procedure for an assessment of failure probability of RPV as well as of primary piping including collection and analysis of all necessary materials data

RPV related Projects :

- Guidance for the Residual Lifetime Evaluation of Reactor Presure Vessels with Cladding
- Procedure for an assessment of defects in a cladding of WWER-440 RPVs

- Re-evaluation of a standard surveillance programme for WWER-440/V-213 RPVs
- Design and realization of a supplementary surveillance programme for WWER-440/V-213 RPVs

- Measurement of "instrumented hardness identation" of cladding during ISI

- Procedure and evaluation of annealing efficiency and residual lifetime of WWER-440/V-230 RPVs in NPP Jaslovské Bohunice
- Assesment of residual lifetime of RPVs and their internals of WWER-440/V-213 reactors in NPP Dukovany
- Procedure for an assessment of lifetime of RPVs and internals of WWER-1000/V-320 reactors in NPP Temelín

- Cooperation in the IAEA CRP on RPV materials irradiation embrittlement, Phase III
- Radiation damage mechanisms study in WWER RPVs materials
- Irradiation embrittlement study of RPVs materials including cladding

- Study of the behaviour of WWER RPVs materials under different corrosion conditions including a simultaneous effect of reactor irradiation

- Cooperation in the IAEA Pilot Study on RPV primary nozzle ageing of WWER-440/V-213 type

- Ascertainability of real defects in samples of main components of primary circuits of WWER-440 and WWER-1000 using ultrasonics detection technics

LEAK-BEFORE-BREAK PROJECTS

DESIGN BASIS ASSESMENT OF LBB APPLICABILITY TO SAFETY RELATED PIPINGS IN PRIMARY CIRCUITS

- NPP JASLOVSKÉ BOHUNICE
  - WWER-440/V-230
  - WWER-440/V-213
- NPP MOCHOVCE
  - WWER-440/V-213
- NPP DUKOVANY
  - WWER-440/V-213
- NPP TEMELÍN
  - WWER-1000/V-320

DESIGN BASIS ASSESSMENT OF 0.1 F LEAK-BEFORE-BREAK APPLICABILITY

- NPP JASLOVSKÉ BOHUNICE
  - WWER-440/V-230

0.1 F means a leak from an opened through-crack with a surface equal to 0.1 of a piping cross section

for example : 0.1 F from ID 500 mm is equivalent to DGB of piping with ID 200 mm

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Rules for an assessment of RPV lifetime and defects allowability

MAIN AIMS :

- to prepare Rules based on PWR principles applied to Soviet WWER design and materials taking into account existence of austenitic cladding

MAIN DIRECTIONS :

- comparison of Soviet and ASME/US NRC Codes

- comparison of Soviet and ASME/US NRC calculation and testing procedures and methods

- supplementary tests of mechanical properties according to ASME and ASTM standards and requirements - RT_{NDT}, K_{Id}, K_{IA} for WWER-440 and WWER-1000 materials-

- determination of the effect of :
  - biaxial stress state on K_I as well as design fracture toughness curves
  - cladding presence of K_I values for surface and sub-surface defects
  - effect of residual stresses in cladding on its behaviour
  - effect of small defects on design fracture toughness curves

- supplementary irradiation tests of cladding materials

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Appendix I

PROCEDURE ON DEFECTS ACCEPTANCE EVALUATION FOR APPLIANCES IN PRIMARY CIRCUIT OF NPP V-213

Procedure layout

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Appendix II

PROCEDURE ON THE EVALUATION OF STANDARD AND SUPPLEMENTARY PROGRAMME WITH SURVEILLANCE SPECIMENS

Procedure layout

Introduction

1. Validity limitations
2. General provisions
3. Requirements on standard and supplementary surveillance programme
   - materials selection
   - monitoring of neutron fluxes and fluences within the programme on RPV
   - monitoring of irradiation temperature
   - types of mechanical tests
4. Procedure of neutron fluence evaluation
   - particular tests
   - particular containers
   - set of containers for one set of specimens
   - complete set of containers
5. Procedure on irradiation temperature evaluation
   - monitors of diamond type
   - melting monitors
6. Procedure of mechanical tests and their evaluation
   - standard tensile tests
   - standard impact energy test (with record)
   - standard fracture toughness test
   - standard HV-hardness test
   - reconstitution of irradiated specimens
7. Methodology of results analysis from particular sets of containers
   - from the point of view of neutron fluence
   - from the point of view of irradiation temperature
   - from the point of view of mechanical tests results
8. Comparison of design and real radiation damage of RPV materials
9. Way of interpretation of particular container sets and their sets from particular reactors for residual lifetime evaluation

   Attachment A - Methodology for cladding degrading evaluation by the "kinematic hardness" method

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Appendix III

EVALUATION RADIATION EXPOSURE ON PRESSURE VESSELS OF REACTOR TYPE V-213

Procedure layout

Appendix IV

PROBABILISTIC EVALUATION OF FAILURE RESISTANCE OF RPV

Procedure layout

1.0 Definition of failure criteria
  1.1 Criteria of RPV failure
2.0 Postulated loading regimes
  2.1 Regimes significant for RPV
    2.1.1 Regimes violating normal operation conditions
    2.1.2 Accidental regimes (transient PTS)
3.0 Fracture mechanics parameters for load-carrying capacity evaluation
  3.1 Fracture mechanics parameters used for RPV evaluation
    3.1.1 Conditions on defects distribution
    3.1.2 Limitations valid for probability of defects detection and for establishing of critical size
    3.1.3 Requirements on next operation parameters
      3.1.3.1 Covering operation inspections
      3.1.3.2 Covering of residual stresses
      3.1.3.3 Overload influence estimation at temperature
4.0 Definition and assessment of socially acceptable risk
  4.1 Criterion of socially acceptable risk
  4.2 Compatibility of deterministic and probabilistic evaluation - deterministic formulation of criterion of obligatory probabilistic evaluation
  4.3 Socially acceptable risk - evaluation in probabilistic terms

Current limitations and shortcomings in probabilistic assessments of brittle failure are caused by lack of data about:

- distribution of characteristic defects and their:
  - location
  - characteristic size
- fracture toughness at crack arrest in case of reactor pressure vessel
- statistical fracture toughness
- probability of defect detection
- uncertainty in establishment of phosphorus and copper content
- inaccuracy of establishment of real fluences in case of the evaluation of resistance against failure of the RPV

The simplified distribution of characteristic defects refers usually, due to lack of data, to one-dimensional distribution of defect depths but not defect lengths.

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Appendix E-1

INFLUENCE OF AUSTENITIC CLADDING ON THE STRESS FIELD AND FRACTURE MECHANICS PARAMETERS FOR THE LOCATION OF AUSTENITE/FERRITE TRANSITION

Objective: Establishment of correction factors for fracture mechanics parameters calculation in case of defects located in cladding region (cladding, cladding / base material transition, underclad cracks)

Appendix E-2

ESTABLISHMENT OF TENSILE AND FRACTURE PROPERTIES OF CLADDING AND THEIR DEGRADING UNDER OPERATION CONDITIONS

Objective: Establishment of design fracture toughness and trend curves for RPV and assessment of their changes due to operation degrading by use of both standard (destructive) tests and indentation diagrams

Appendix E-3

INFLUENCE OF BIAXIALITY, RELATIVE AND ABSOLUTE CRACK LENGTH AND LOADING MODE ON FRACTURE TOUGHNESS OF MATERIALS OF RPV

Objective: Estimation of correction coefficients allowing for the influence of different properties of cladding and base material, and if needed also of the heterogeneous weld, on cracked region behaviour when loaded to elastic and elastic-plastic level

Appendix E-4

EVALUATION OF THE INFLUENCE OF DIFFERENT PLASTIC PROPERTIES OF CLADDING AND BASE MATERIAL ON THE SINGULAR FIELD ALONG THE CRACK FRONT AND CONSEQUENTLY ON FRACTURE TOUGHNESS.

Objective: Establishment of design curves of fracture toughness (and/or correction factors) for various levels of biaxiality due to both normal operation and thermal shock. Cladding influence must be taken into account. Establishment of correction factors for small defects (both surface and embedded ones) for fracture mechanics parameters computation and for design fracture toughness curves respecting the influence of austenitic cladding

UPGRADING OF VVER-440 REACTOR PRESSURE VESSEL SURVEILLANCE PROGRAMME

STANDARD SURVEILLANCE PROGRAMME :

- duration practically only 5 years
- high lead factor (13-17)
- steep gradient of fast neutron flux
    - in containers chain
    - in container chain position
- wide neutron fluence values in one set of specimens (COD type)
- temperature monitors of high uncertaintity
- uncertaintity in specimen fluence determination in containers

PRINCIPLES OF SUPPLEMENTARY SURVEILLANCE SPECIMEN PROGRAMME

REQUIREMENTS :

- use of real RPV materials
- low lead factor (2-3)
- more detailed measurement of neutron fluence
- measurement of irradiation temperature
- monitoring of fluences and material property changes during whole RPV residual lifetime

RE-EVALUATION OF STANDARD PROGRAMME

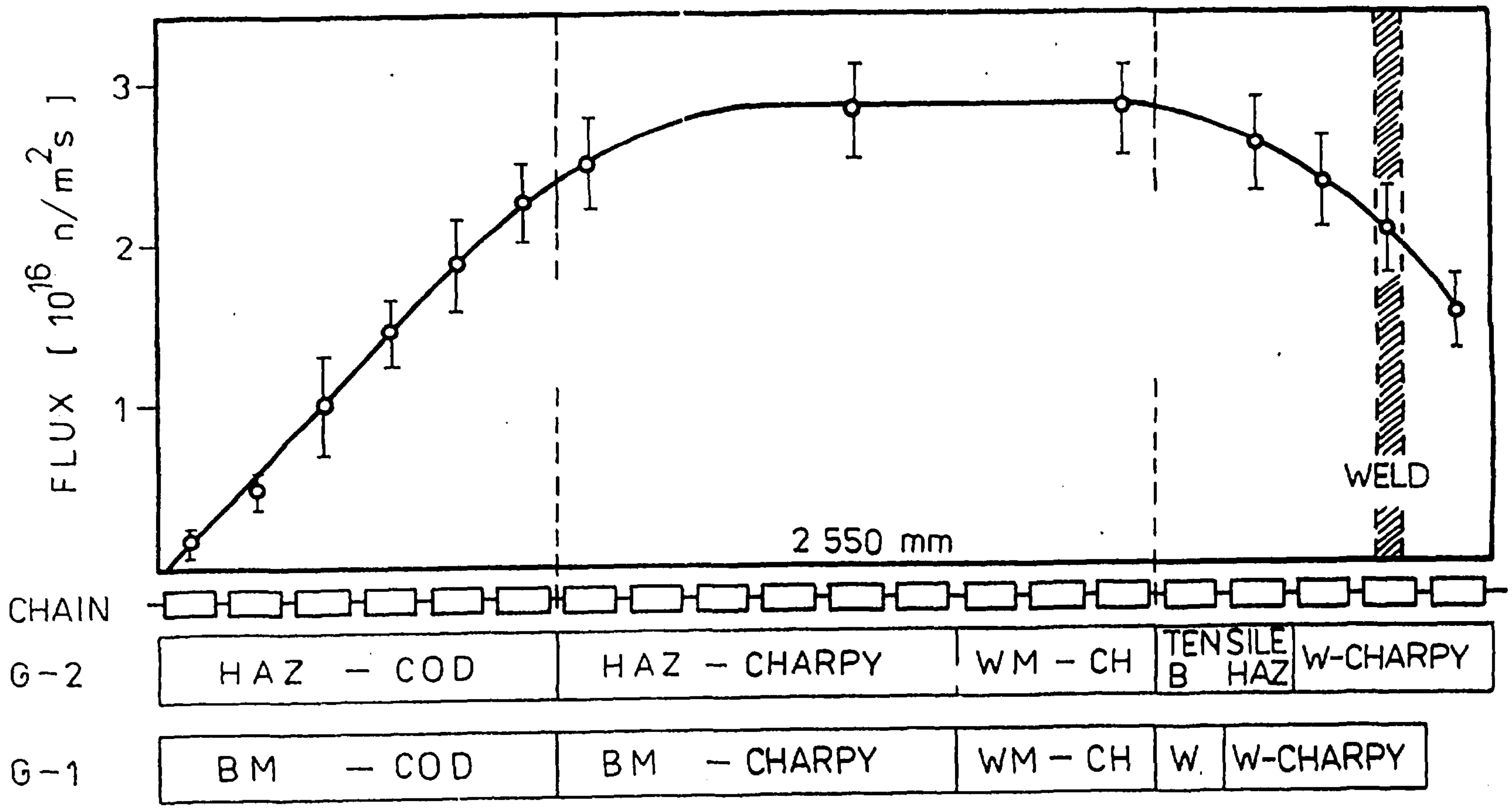
- re-evaluation of neutron fluences
  - use of specimen scanning method
  - re-calculation of neutron fluxes and fluences
- reconstitution of irradiated Charpy-V type specimens
  - static fracture toughness
  - dynamic fracture toughness
- analysis of surveillance test results)
- preparation of trend curves for dT shift

SOLUTION = DESIGN :

- use of specimen reconstitution method (including reference one)
- new design of irradiation containers
  - temperature monitors
  - fluence monitors
  - 3x5 specimens of 10x10x14 mm
- containers placing into low neutron fluence
- wider use of fluence monitors (including fission ones)
- connection between surveillance and ex-vessel fluence monitoring
- type of tests:
  - tensile
  - instrumented Charpy-V notch toughness
  - static fracture toughness on pre-cracked Charpy
  - dynamic fracture toughness on pre-cracked Charpy

EXPERIMENTAL - COMPUTATION VERIFICATION:

- "measuring chain" in 1993 in EDU NPP
  of standard and supplementary programme containers
  - measurement of fluences and spectra (set of monitors)
    - in-vessel and ex-vessel
  - measurement of irradiation temperatures (thermocouples)
  - measurement of gamma flux (calorimeter)
  - 3D calculations of time dependent fluxes


FLUX [10^{16} n/m^{2}s]
3
2
1
2 550 mm
WELD
CHAIN
G-2
HAZ – COD
HAZ – CHARPY
WM – CH
TENSILE
B HAZ
W-CHARPY
G-1
BM – COD
BM – CHARPY
WM – CH
W.
W-CHARPY

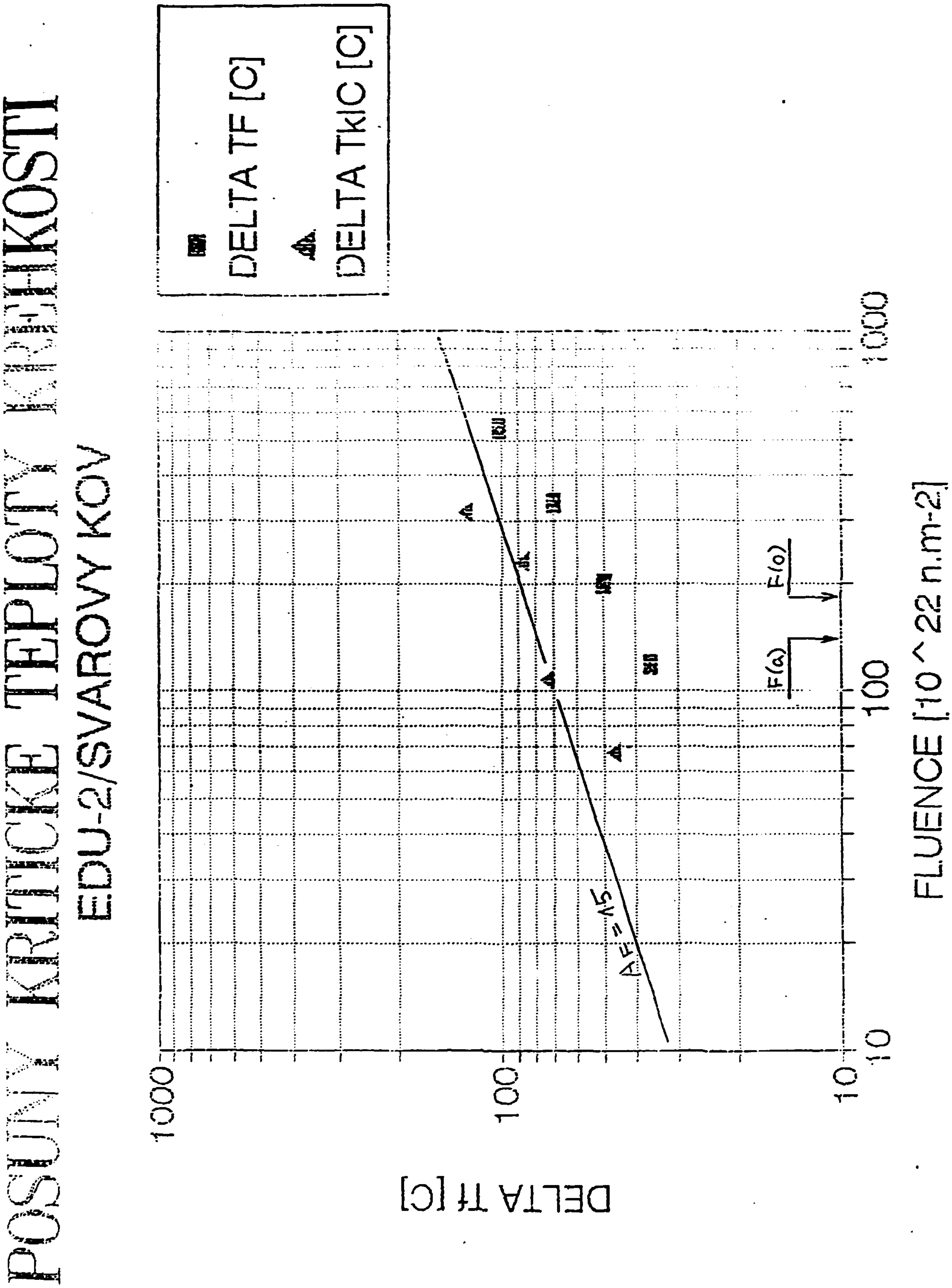

POSUNY KRITICKE TEPLOTY KREHKOSTI
EDU-2/SVAROVY KOV
DELTA TF [C]
DELTA TkIC [C]
DELTA Tf [C]
FLUENCE [10 ^ 22 n.m-2]
1000
100
10
ΔF = 15
F(a)
F(o)

Fig. 3

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19. MPA-SEMINAR STUTTGART

Kinematic Hardness Measurement in Cladding as a Part of Reactor Pressure Vessel in-Service Inspection and Surveillance Programme

M. Brumovský, J. Novák and J. Žďárek

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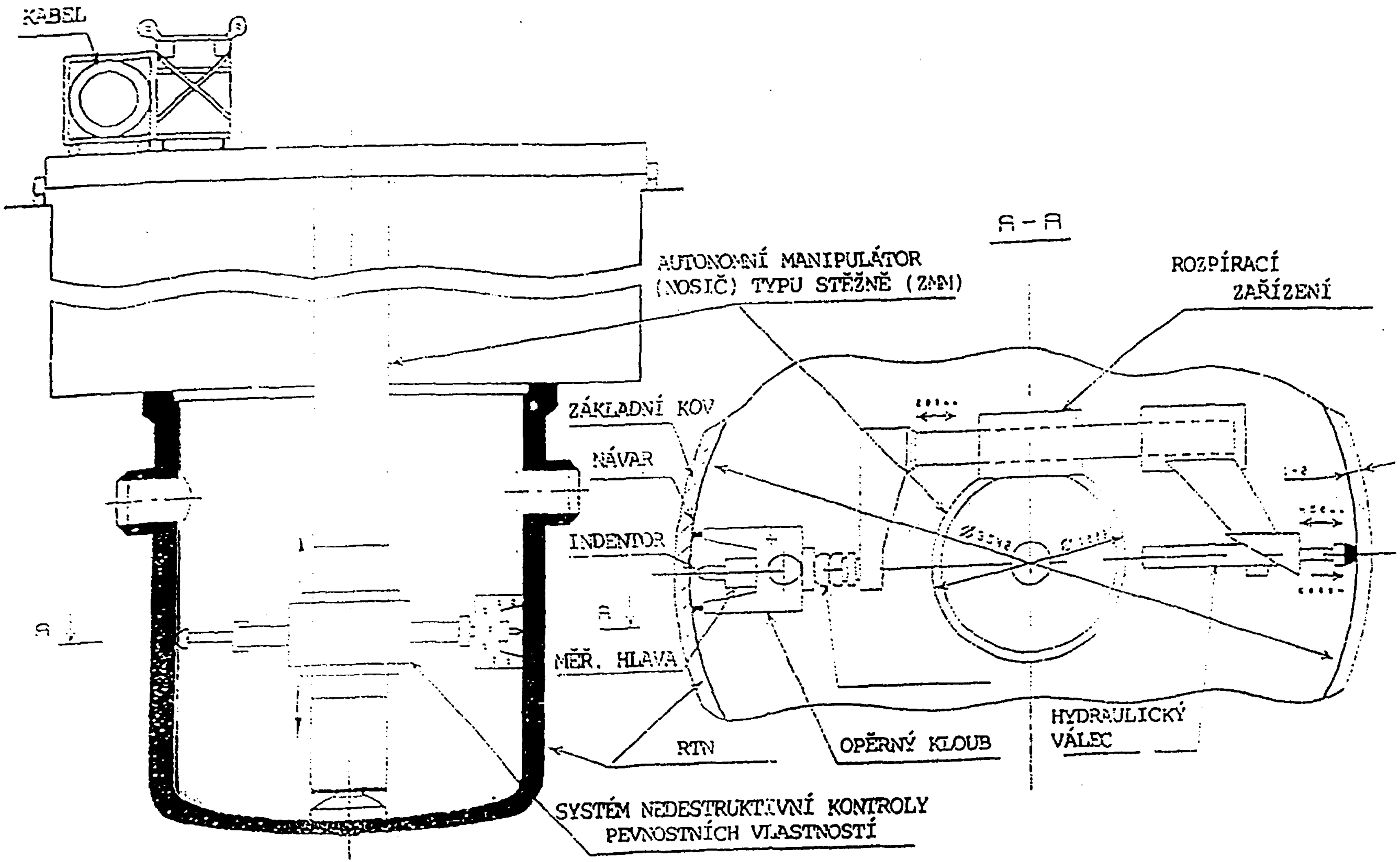
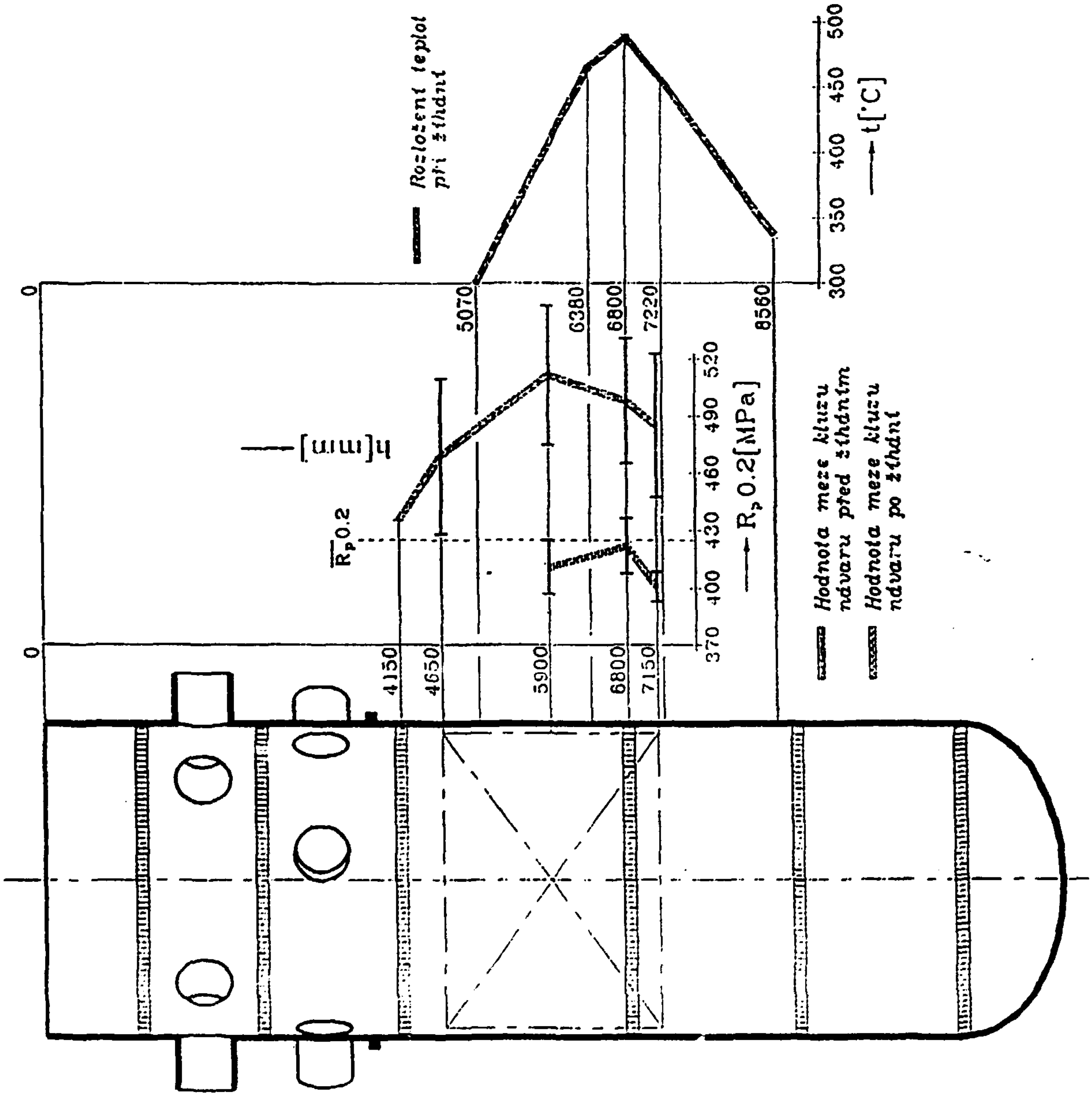
Obr. 1. Zařízení nedestruktivní kontroly pevnostních vlastností materiálu vnitřního povrchu RTN





Obr. 3. Výškový profil meze kluzu návaru $R_p0{,}2$ podle vyhodnocení VNIIAES před i po regeneračním žíhání, pro úhlové pozice 30^{o} a 12.5^{o} spolu s výškovým profilem teplot při žíhání

PRESSURE VESSEL ANNEALING PROGRAMME :

- TECHNOLOGICAL PROGRAMME :
  - choice of annealing heat treatment :
    465 +/- 10 C - 164 hs
  - calculation of stress-temperature fields in RPV, optimation of temperature field
  - design and manufacturing of annealing furnace
  - manufacturing of a RPV model in scale 1:1
  - test annealing of this model with furnace in shop(ŠKODA)
  - measurement of temperatures, stresses and residual stresses

- IRRADIATION EXPERIMENTS PROGRAMME :
  - irradiation of "quasi-archive" materials in V-213 host reactor in surveillance positions (see Table "Irradiation-annealing combinations")
  - irradiation (up to 3 cycles) in experimental reactor LVR-15 (NRI) of "tailored" weld metal
  - irradiation (one cycle) in V-213 host reactor of "tailored" weld metal
  - mechanical testing of irradiated and annealed materials
  - thermal ageing - time schedule equivalent to a proposed annealing heat treatment

LARGE SCALE TESTS :
- fracture toughness testing
- PTS large scale experiments on artifically aged materials

RPV MATERIAL VERIFICATION PROGRAMME :

- precisioning of chemical composition
- precisioning of actual neutron fluences (cavity measurements)
- measurement of deformation diagramme during hardness testing :
    - base metal - specimens from outer surface
    - cladding - measurement on inside surface

RPV LIFETIME RE-ASSESSMENT :

- re-calculation of lifetime assessment, using :
    - precisioned chemical composition
    - precisioned mechanical properties
    - precisioned irradiation embrittlement coefficient
    - precisioned neutron fluences
    - new data on material defectness (ISI before and after annealing)
    - precisioned water temperature regimes, temperature-stress fields in RPV
    - verification of fracture toughness curve from experimental testin analysis
    - checking of the possibility of use of crack-arrest approach (large scale PTS experiments)

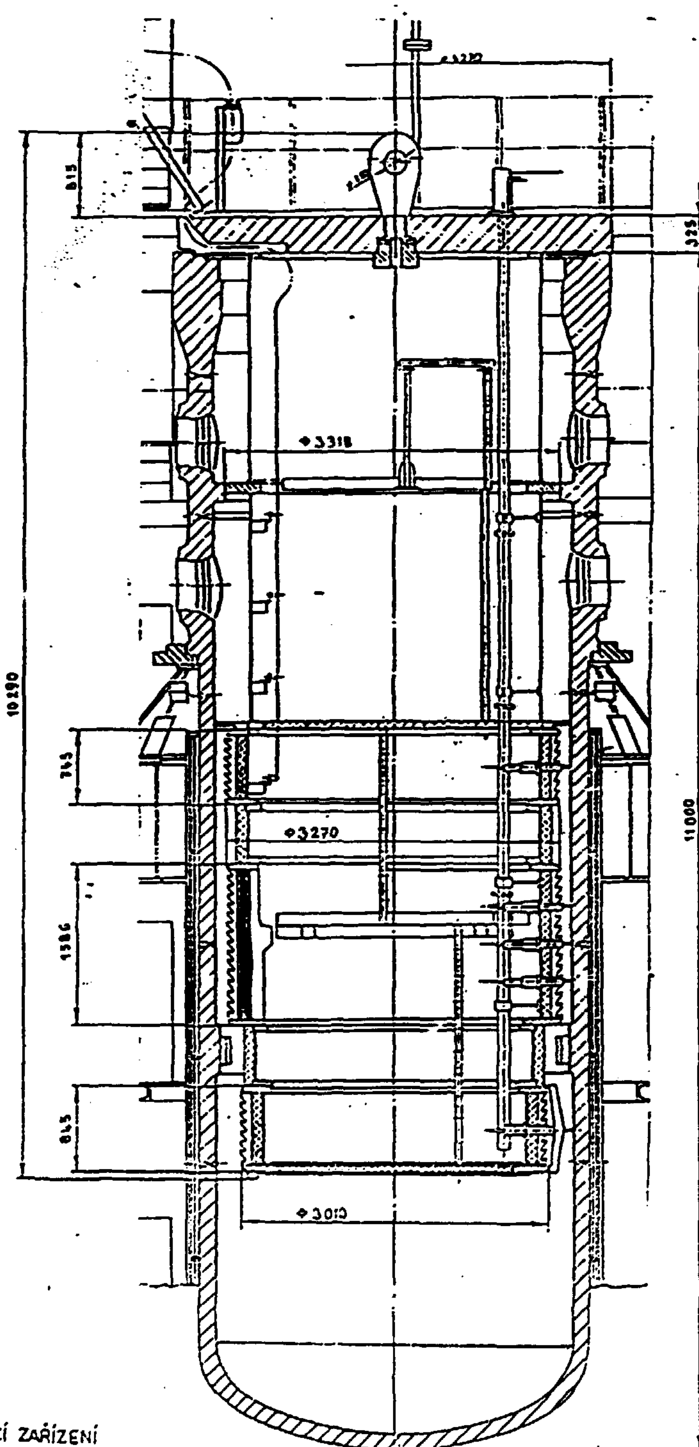
PROBABILISTIC FRACTURE MECHANICS CALCULATIONS :

SEE - PAPER OF Dr. HORÁČEK (ŠKODA)

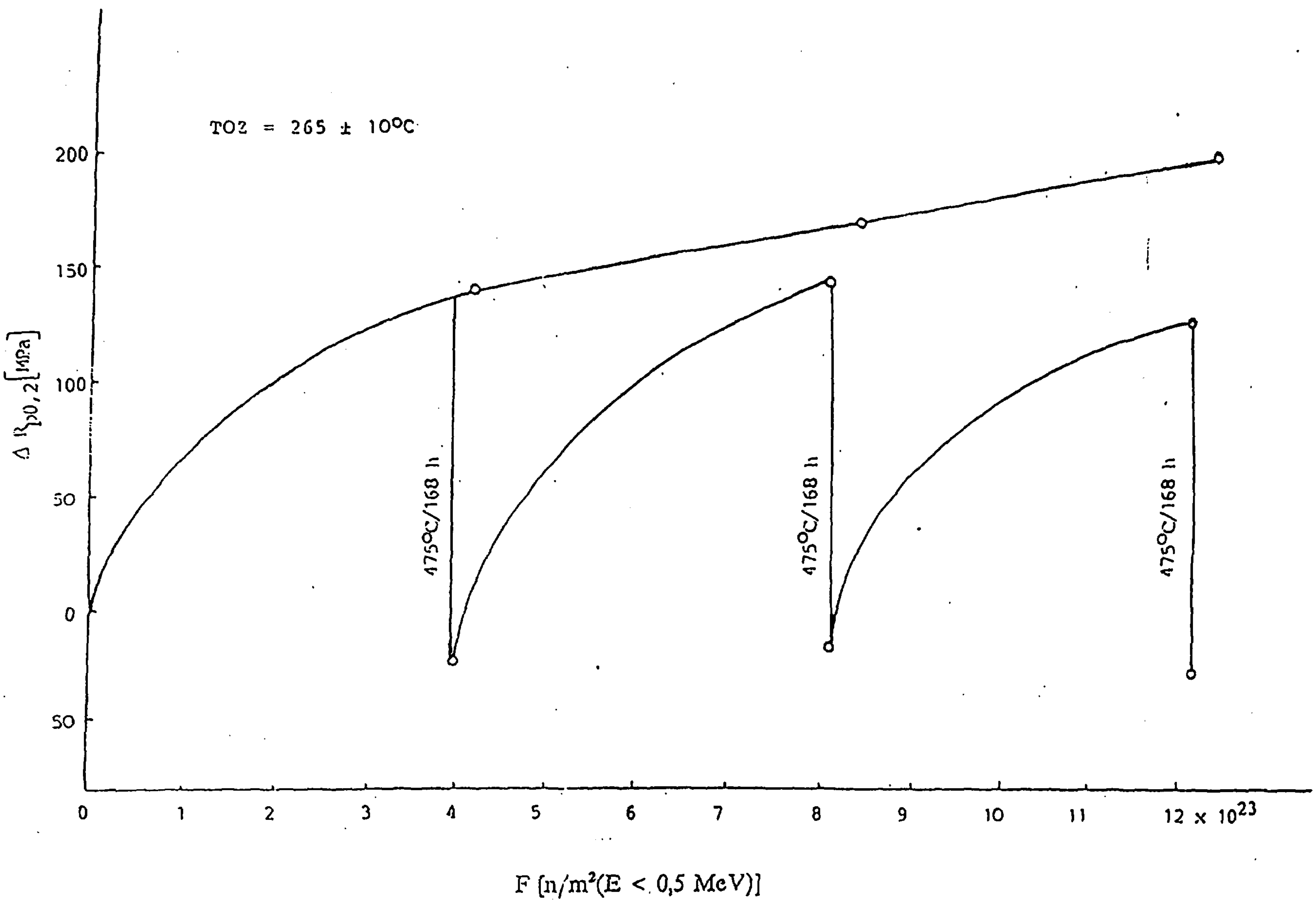
IRRADIATION-ANNEALING COMBINATIONS

MATERIAL	TEST	REGIME U	I	I+A	I+A+I	I+A+I+A	2I	2I+A	A
WM-	T	*	*	*	*	*	*	*	+
	KCV	*	*	*	*	*	*	*	+
	K_{IC}	*	*	*	*	*	*	*	+
	K_{Id}	*	*	*					
	HV	*	*	*	*	*	*	*	+
WM-Root	T	*	*	*					
	KCV	*	*	*					+
	K_{IC}	*	*	*					
	HV	*	*	*					
BM-	T	*	*	*	*	*			+
	KCV	*	*	*	*	*			+
	K_{IC}	*	*	*	*	*			+
	K_{Id}	*	*	*					
	HV	*	*	*	*	*			+
Cladding	T	*	*	*					+
	K_{IC}	*	*	*					
	K_{Id}	*	*	*					
IAEA-1	KCV	*	*	*					
IAEA-2	KCV	*	*	*					

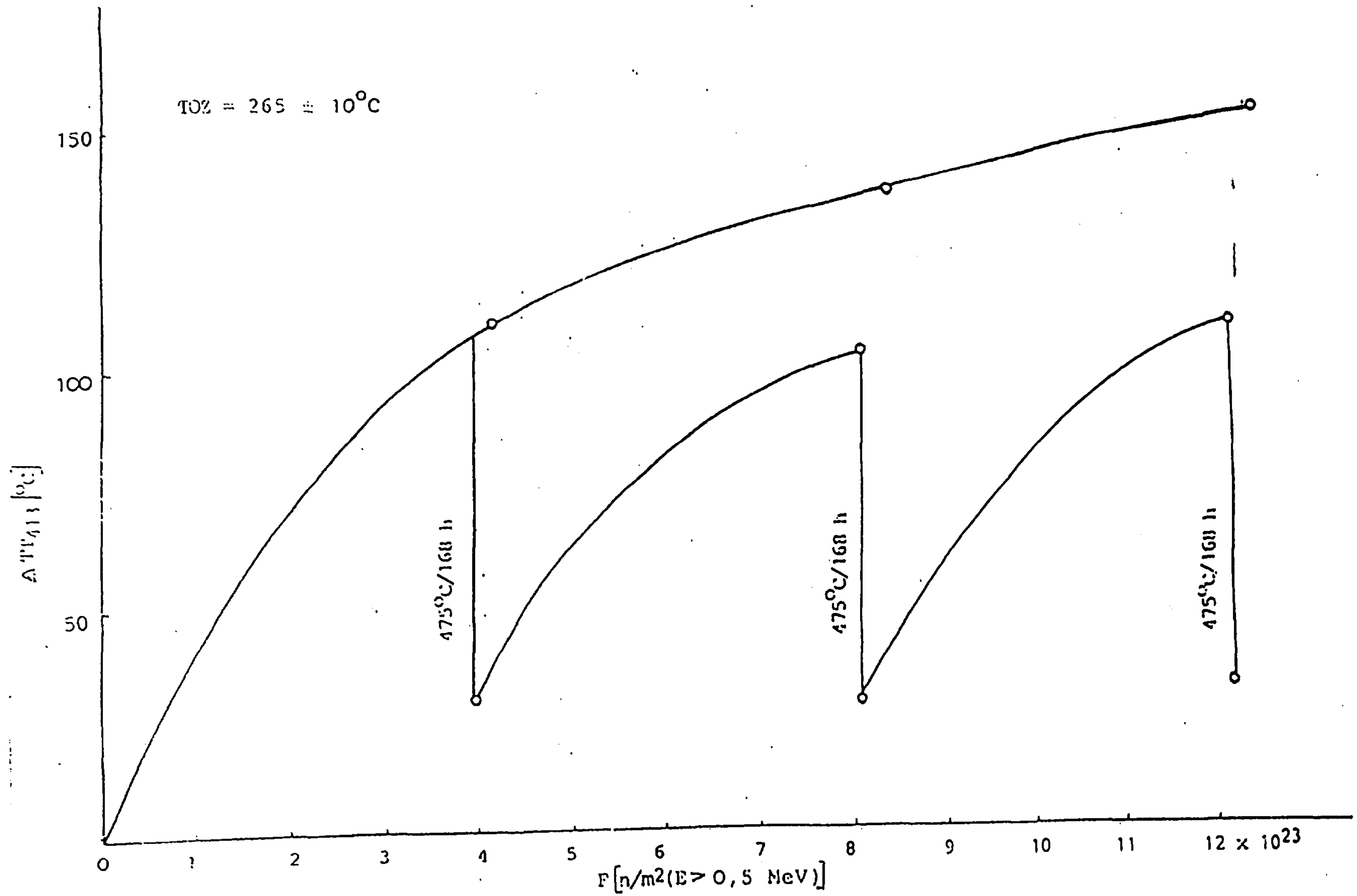
WM - WELD METAL
WM-Root - WELD METAL - ROOT
BM - BASE METAL
Cladding - CLADDING METAL
IAEA - IAEA COORDINATED RESEARCH PROGRAMME MATERIALS



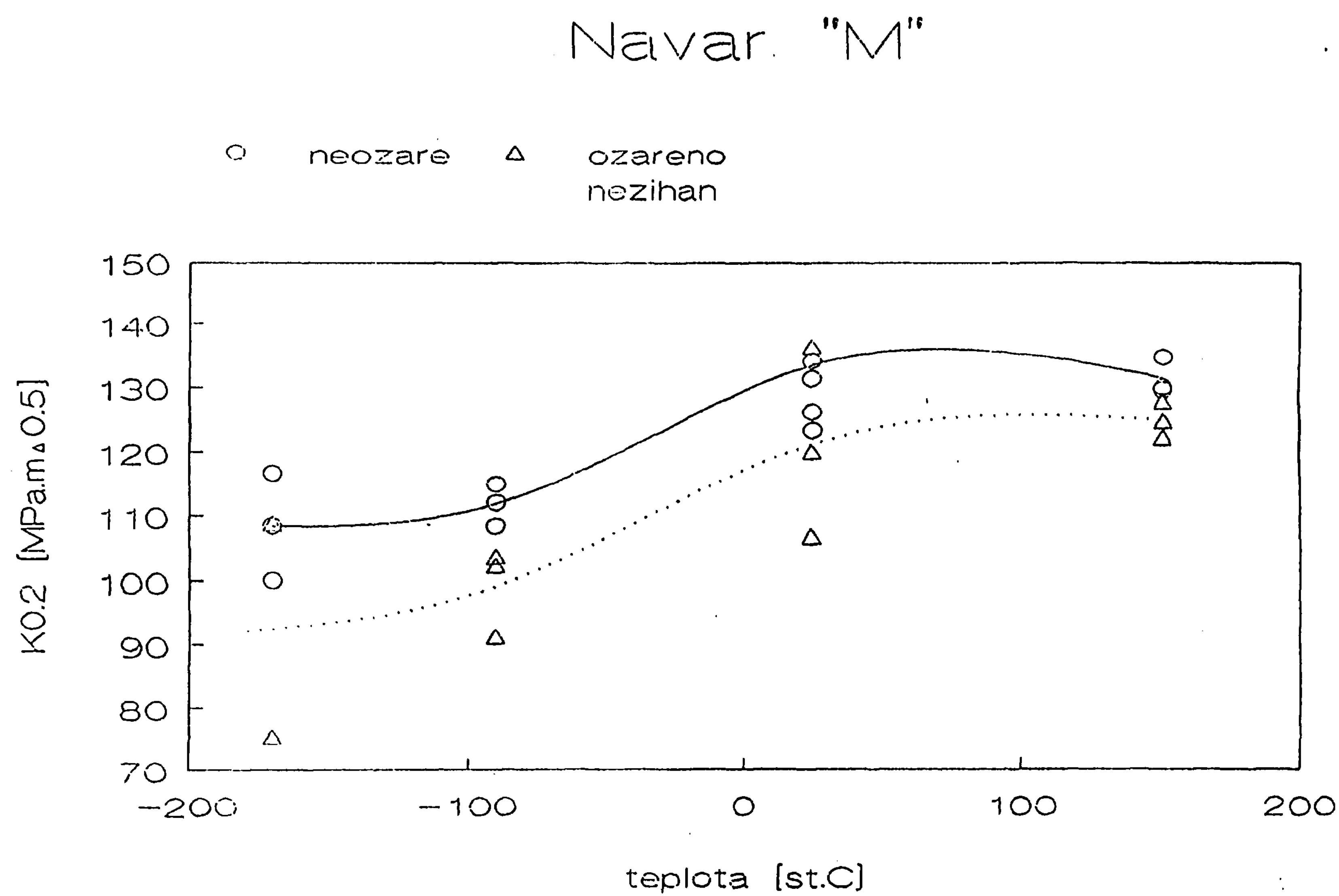
ŽÍHACÍ ZAŘÍZENÍ



Obr. 4: Závislost radiačního zpevnění ($\Delta R_{p0,2}$) a stupně zotavení na neutronové fluenci a aplikovaném regeneračním žíhání pro SK K.



Obr. 6 Závislost radiačního zkřehnutí a stupně zotavení na neutronové fluenci a aplikovaném regeneračním žíhání



Obr. 22. Závislost lomové houževnatosti neozářeného a ozářeného (5,6 x $10^{23} m^{-2}$, E>0,5 MeV) návaru "M" na zkušební teplotě.

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EXPERIMENTAL FOR SSRT PROCEDURE

1.Pre-irradiation of SSRT specimens
- Research reactor LWR-15
- Dry irradiation facility:
  Chouca - MT rig (French production)
  specimens block

temperature range	290-300°C
inert atmosphere	He
fast neutrom flux	1.7×10^{17} $nm^{-2}s^{-1}$ (E>1MeV)
fast neutron fluence	up to 5×10^{24} nm^{-2} (E>1MeV)

2.Preparation of SSRT in hot cell
- dismounting of dry irradiation rig
- removal of specimens
- insertion of the test specimen into SSRT grips

3.SSRT experiment in pile channel of reactor water loop
- temperature 290 - 300°C, pressure 12.5 MPa, neutron fluence > 5×10^{24} nm^{-2}
- maximum load in tension 25 kN
  strain rate 10^{-6} mm/s
- maximum displacement 16 mm
- PWR and BWR water chemistry

4.Evaluation
- stress corrosion crack rate
- crack initiation stress and strain
- scanning electron microscopy
- microanalysis
- fractography

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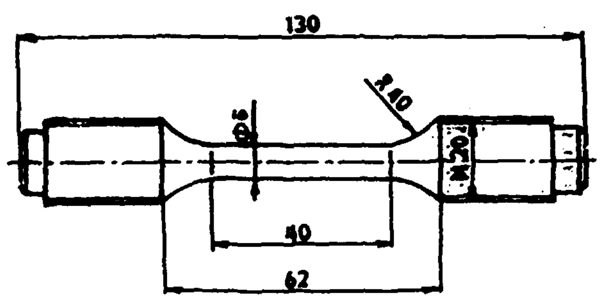
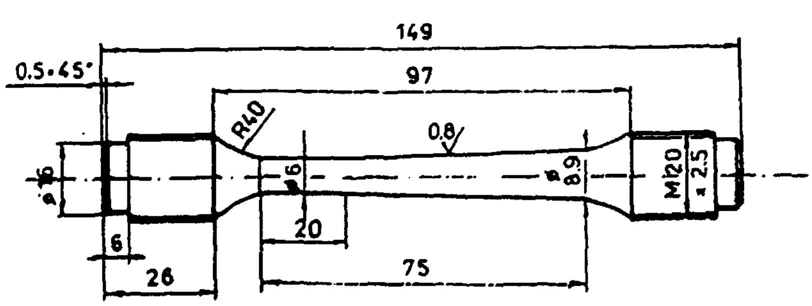
MATERIALS

Chemical composition (in wt%)

	C	Mn	P	S	Cr	Ni	Ti
08Ch18N10T	0.07	1.28	0.025	0.007	17.7	10.56	0.38
14Ch17N2	0.17	0.49	0.019	0.016	16.5	2.09	-



Cylindrical tensile specimen



Tapered tensile specimen

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REACTOR WATER LOOP MAX. PARAMETERS

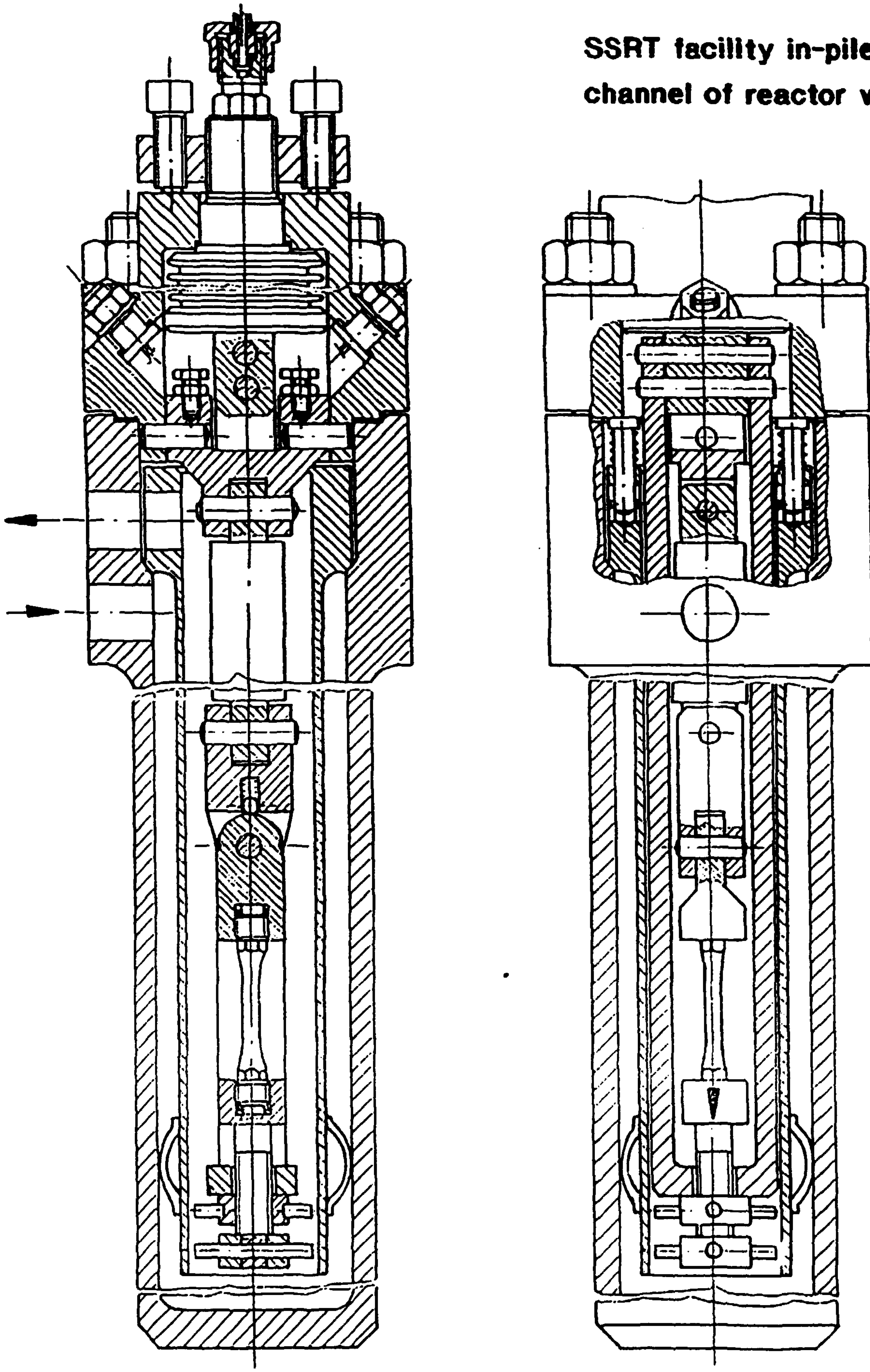
PRESSURE	17 MPa
TEMPERATURE	334 °C
WATER FLOW RATE	10t/hr
ELECTRICAL HEATING	100 kW
COOLING CAPACITY	50 kW
PRIMARY WATER VOLUME	300 dm^3

IRRADIATION PARAMETERS OF THE CHANNELS

Integral neutron flux densities in the centre of loop

O_{fast}	1.74 E+17 n/m^2s
$O_{epith.}$	1.2 E+17 n/m^2s
$O_{therm.}$	1.32 E+17 n/m^2s
O_{gamma}	6.5 E+18 n/m^2s

SSRT facility in-pile channel of reactor water loop



"Ascertainability of true defects in the samples of main components of primary circuits of the VVER 440 and VVER 1000 NPPs"

Project sponsor: State Office for Nuclear Safety (SONS)
Realization: Nuclear Research Institute Rez, plc (NRI)
Integrity and Material Division

with a cooperation of defectoscopy teams of participating organizations

Main goals of the project:

Evaluation of ascertainability of true defects in the most important materials of chosen main components of primary circuits of the VVER 440 and VVER 1000 NPP in consideration of their size and locality.

Review of true possibilities of defectoscopy teams and applied apparatus.

Obtained results will be evaluated and its analysis will be performed for a need of the SONS in order to review a level of performed ultrasonic defectoscopy control of nuclear equipment.

On the base of evaluation the results and analysis, a probability estimation of defects ascertainability in the welded joints will be performed, and principle of the probability evaluation of welded joint ascertainability will be set.

Evaluation and analysis will be performed from the point of view of
- applied defectoscopy equipment
- participating defectoscopy teams
- types and localization of defects
- applied probes and methodology
- time of real working capacity and efficiency

Description of test specimen for ultrasonic tests

5 test specimens were loaned from the Russia Federation to NRI for this project. Both true surface and space defects typical for manufacture and operation of nuclear equipment (inclusion, pores, lack of fusion, cracks, undercladding cracks).

1) a plate with one side weld joint
2) a plate with both sided weld joint
3) a disc with austenitic clad
4) a pipe with two circumferential welds
5) a pipe segment with repaired welded joint

These specimens are typical geometric models of nuclear power components:

- plates
- pipes

From a material point of view, both base material and filler metal of main components are included:

- Reactor vessel of the VVER 440 and VVER 1000
- Main circulating pipe of primary circuit of NPP's with VVER 440 and VVER 1000 reactors
- Austenitic cladding

● LBB APPROACH:

- 10 CFR Part 50, Appendix A
- NUREG 1061, Volume 3
- USNRC - STANDARD REVIEW PLAN
  3.6.3 LBB EVALUATION PROCEDURES
- COUNTERPARTS OF CZECH NRC

PRINCIPAL FEATURES:

- POSTULATED TWC OF 38 L/MIN
  (MARGIN 10)
- $L_{COLLAPSE} / L_{LEAK} \geq 2$
- MARGIN ON LOAD
  OF UNSTABLE CRACK GROWTH ... 1.4

PROCEDURES:

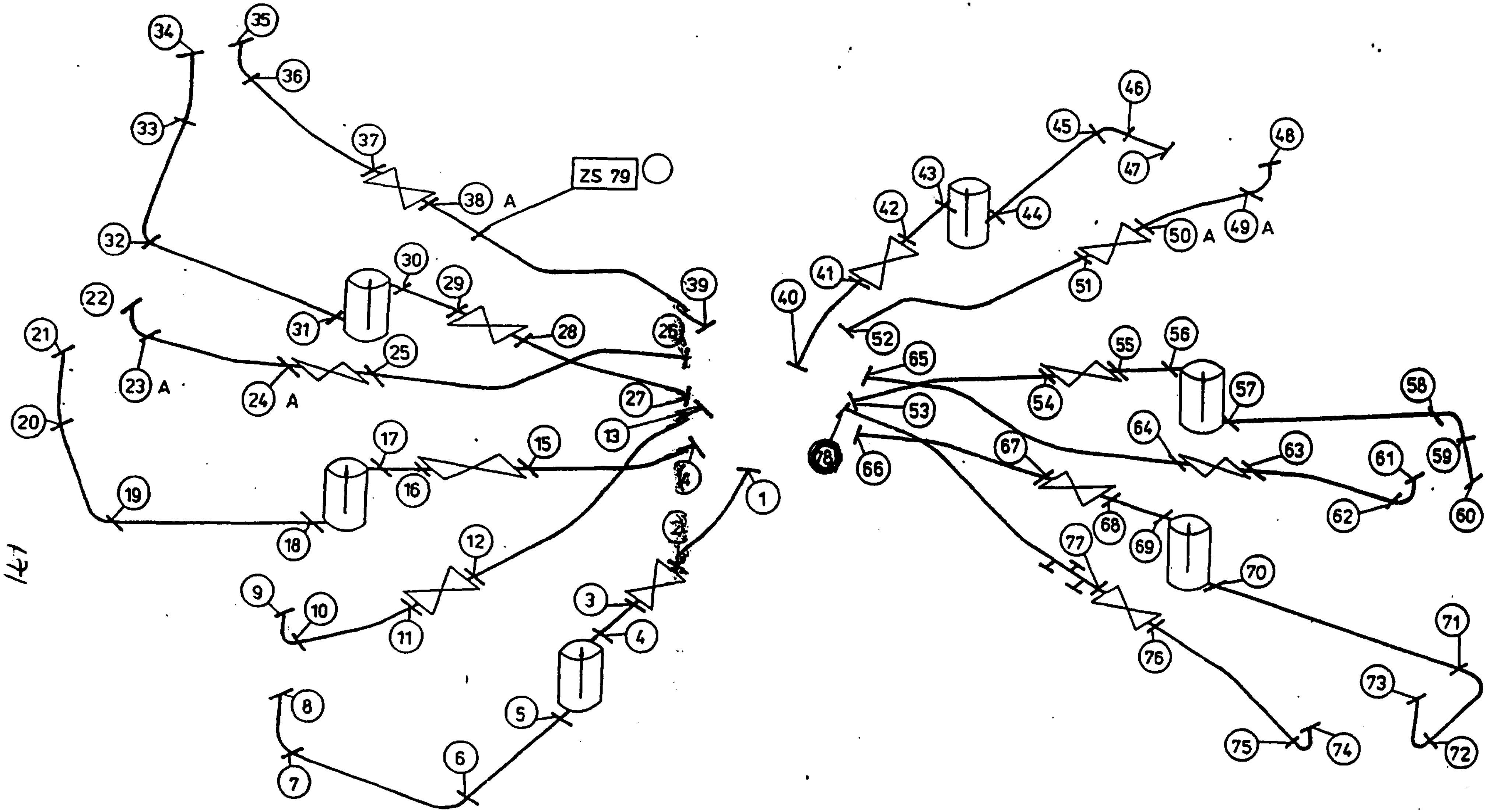
- HANDBOOKS OF SIF AND LL
  (AND/OR FEM PACKAGE SYSTUS)
- R6, J-T, MPA/KWU

● LBB PROJECT LAYOUT:

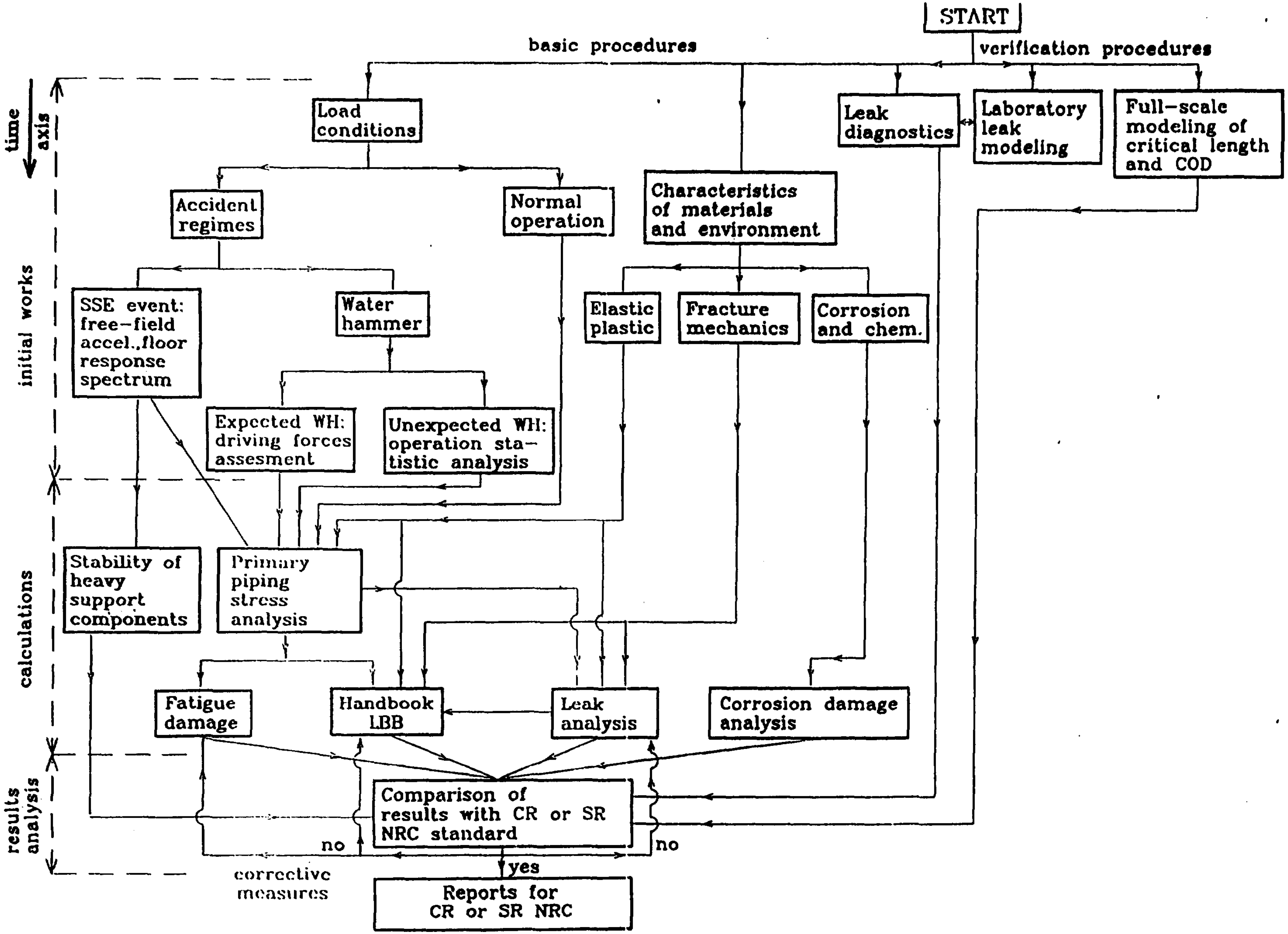
- BASIC PROGRAMME
  - HANDBOOK LBB
  - NEGLIGIBLE CORROSION
  - NEGLIGIBLE FATIGUE
  - LBB SAFETY CASE DOCUMENT

- SUPPORTING PROGRAMME
  - RESPONSE TO NO AND SEISMICITY
  - MATERIAL TESTING

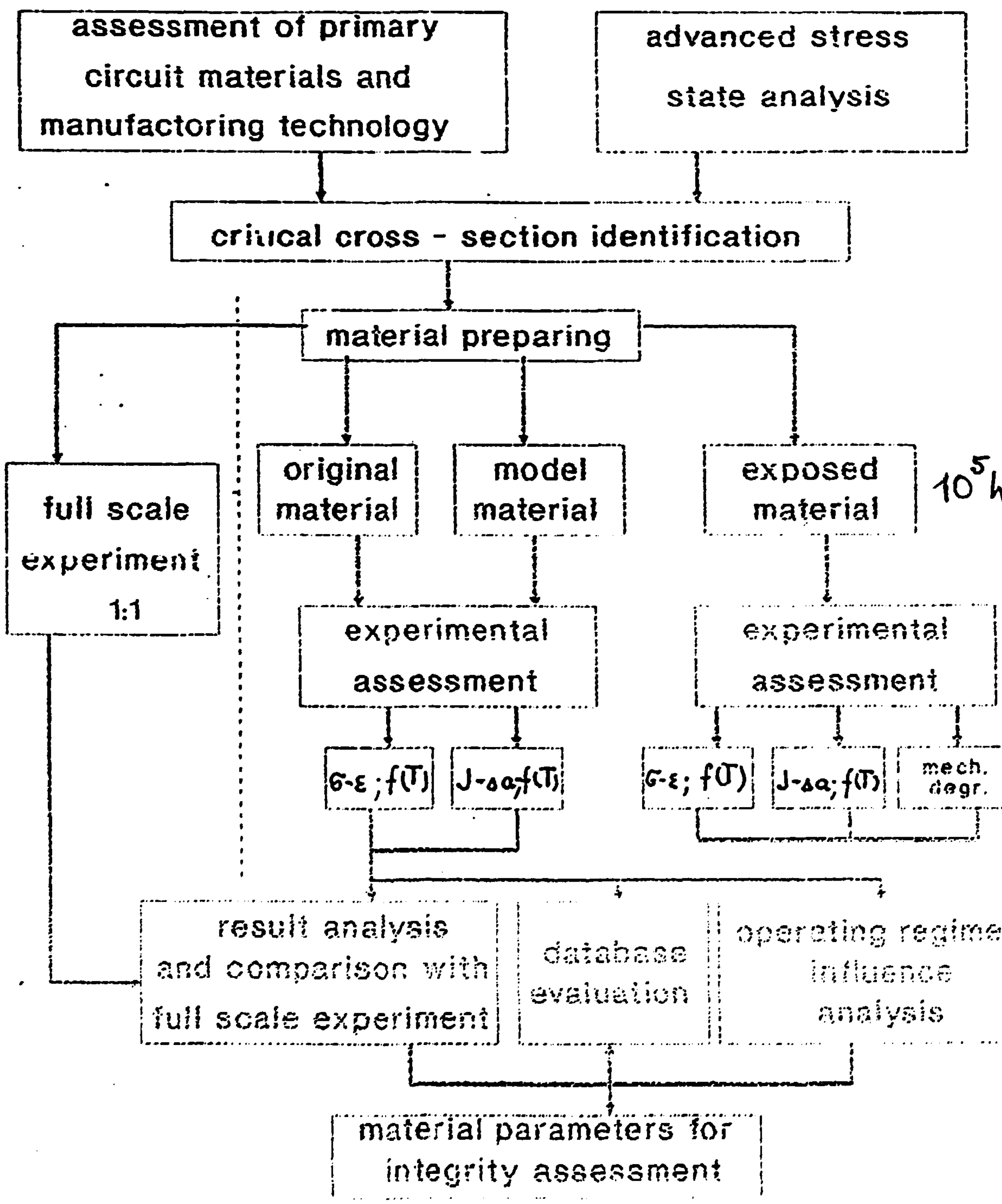
- VERIFICATION PROGRAMME
  - FULL SCALE EXPERIMENTS
    - INTEGRITY PREDICTIONS
    - LEAK PREDICTIONS
    - LEAK DETECTION



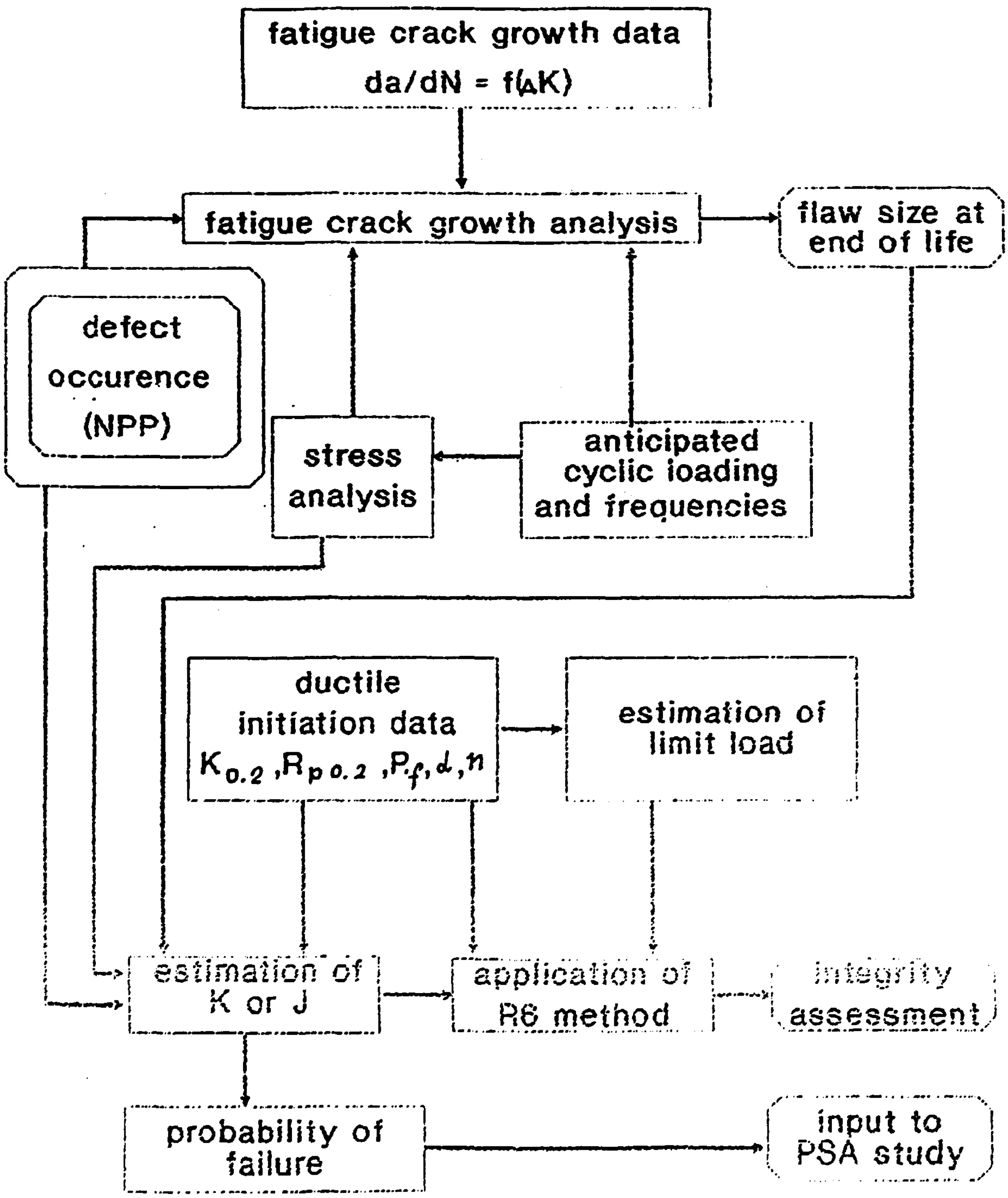
Obr. 3 Orientační schematické znázornění rozložení jednotlivých montážních svarů na hlavním cirkulačním potrubí JE V1.


START
basic procedures
verification procedures
time axis
initial works
calculations
results analysis
Load conditions
Accident regimes
Normal operation
Characteristics of materials and environment
Leak diagnostics
Laboratory leak modeling
Full-scale modeling of critical length and COD
SSE event: free-field accel.,floor response spectrum
Water hammer
Elastic plastic
Fracture mechanics
Corrosion and chem.
Expected WH: driving forces assesment
Unexpected WH: operation statistic analysis
Stability of heavy support components
Primary piping stress analysis
Fatigue damage
Handbook LBB
Leak analysis
Corrosion damage analysis
Comparison of results with CR or SR NRC standard
no
no
yes
corrective measures
Reports for CR or SR NRC

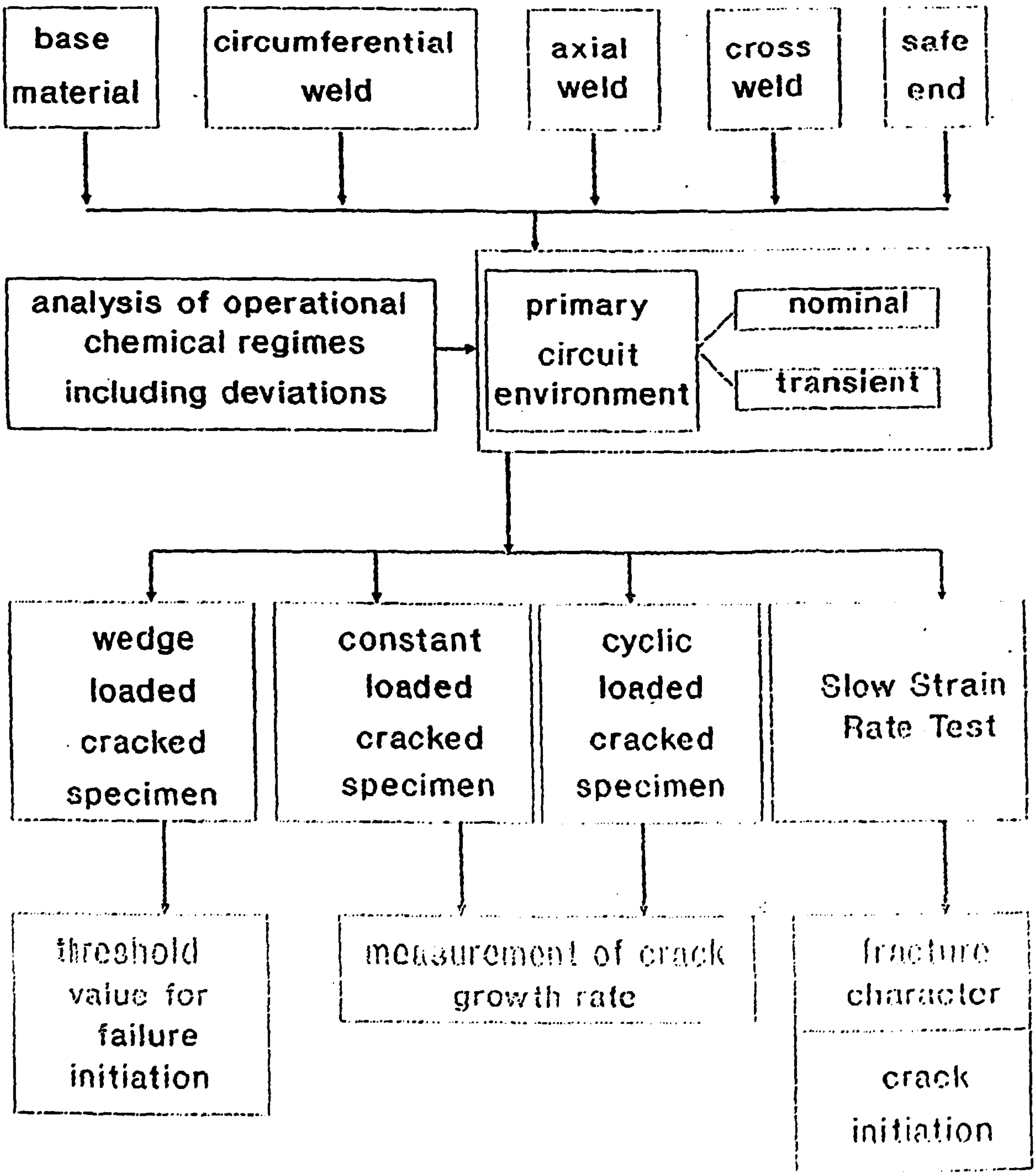
Material characteristics assessment

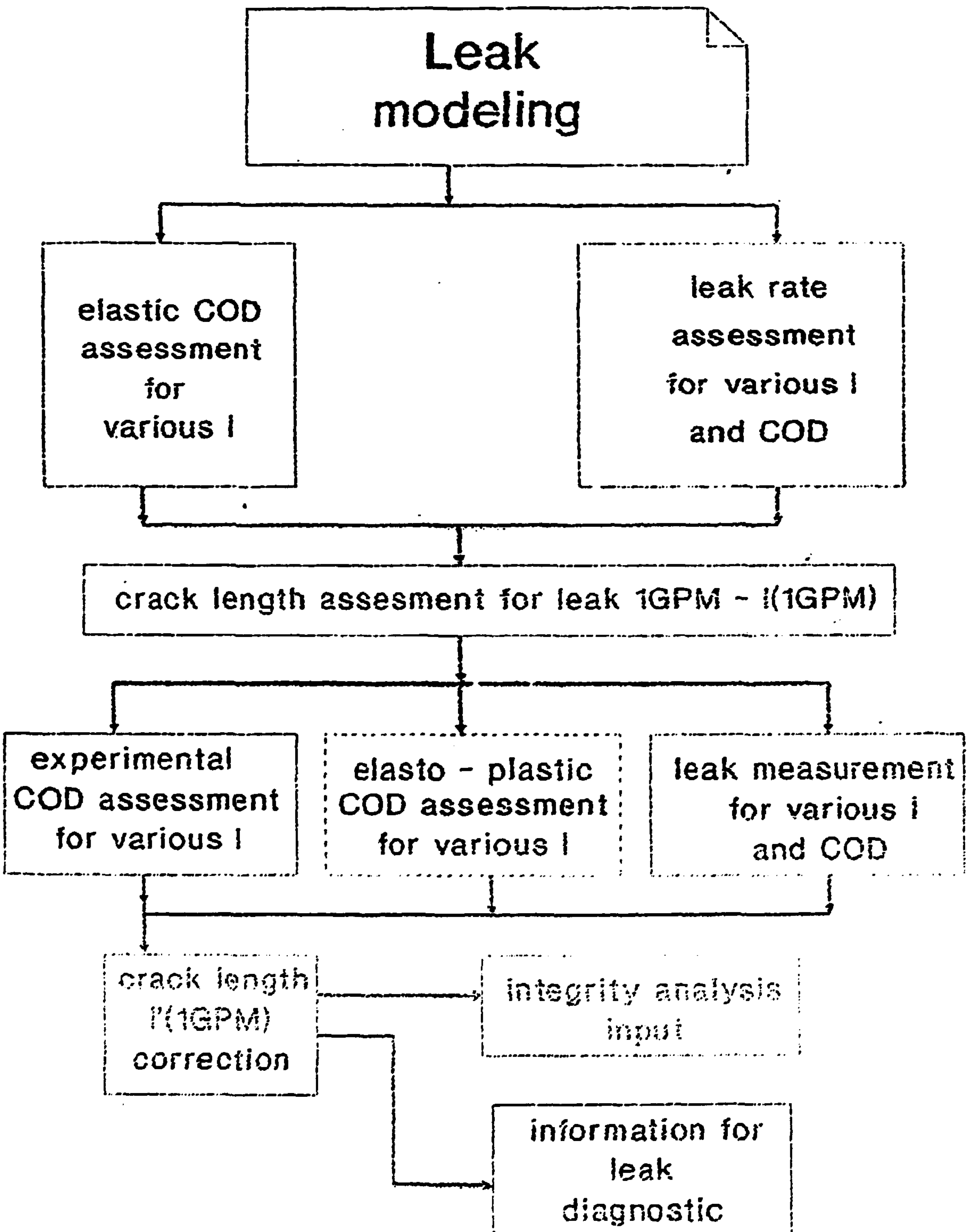


Fracture evaluation



ASSESSMENT OF ENVIRONMENTALLY ASSISTED CRACKING




Leak modeling
elastic COD assessment for various l
leak rate assessment for various l and COD
crack length assesment for leak 1GPM - l(1GPM)
experimental COD assessment for various l
elasto - plastic COD assessment for various l
leak measurement for various l and COD
crack length l'(1GPM) correction
integrity analysis input
information for leak diagnostic

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The LBB Project Conclusions

As a part of the LBB methodology application to the all models of WWER type reactors (e.g. 230, 213 and 320) the static and seismic response analysis had been performed. After assessment of five NPP (V1 and V2 Jaslovské Bohunice, Mochovce, Dukovany and Temelín) which differs in upgrading of the safety significant pipings (main circulating pipe, pressurizer surge lines, feed water, steam and low pressure ECCS pipings), our experiences may be formulated in following manner

- for the assessments of the stress state the complex dynamical models, see fig, 1,2,3 is recommended
- the use of decoupled models results for main circulating pipe in overestimating of the dynamic stresses, which may be classified as conservative approach. The increase of the stresses differs case by case, the maximal value is +10 % for model 213 and 3 % for model 320
- the assessment of seismic responses of uncoupled steam, feed water, low pressure ECCS pipings and pressurizer surge lines results in the rapid underestimation of stresses. The real values from complex models are two or more times higher

Project DBA 0.1 F LBB *

1. Background

International recommendations (SIE, WESI, Russian proposals for backfitting)

CONCEPT OF BACKFITTING OF NPPs` WITH SOVIET TYPE REACTORS WWER 440 MODEL 230

2. Basic concept

- to develope the new safety principle designed as DESIGN BASIS LBB with following basic features

  - for safeends of reactor pressure vessel postulate the circumferential through wall crack with COA equal the cross-section area of pipe with OD 200 mm (pressurizer surge lines)

  - for pressurizer safeends and tees of pressurizer surge lines postulate the same one but with COA equal the 0.1 cross-section area of pipe with OD 200 mm

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* Explanation of terms in text

3. Methodology

3.1. Scope

To verify

- the stability of postulated circumferential through-wall cracks (TWC) for following loading combinations

  - normal operation conditions
  - safe shut-down earthquake with intensity of 8• MSK 64
  - thermal stratification

- the influence of postulated TWC represented as plastic hinges on the redistribution of stresses in piping weldments and in the final step assessment up to the LBB Handbook

- stability of RPV, pressurizer and hot motor operated isolation valve (MOIV) support elements under the loading of TWC jet flow

- the influence of new blow-down conditions on the core barrel stress state

- the stability of electrical cables in the harsh environmental conditions