CLINCH RIVER

BREEDER REACTOR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

VOLUME 13

PROJECT MANAGEMENT CORPORATION

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BREEDER REACTOR PROJECT

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APPENDIX A

COMPUTER CODES

PROJECT MANAGEMENT CORPORATION

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APPENDIX A

COMPUTER CODES

This Appendix is a compilation of brief abstract descriptions of the computer codes used in the evaluation of the CRBRP. Where the code or a particular modification or maintained version of the codes listed below is a proprietary item of the Contractor, it is so noted in the abstract description.

1. AID
2. ANISN-W
3. ANSYS
4. APPROPOS
5. APSA
6. ASHSD2
7. AUTOTEM-III
8. CACECO
8A. CATFISH
9. CHERN
10. CINDA-3G
11. COBRA
12. COMRADEX III
13. CONLIFE
13A. CORINTH
14. COTEC
15. CRAB
16. CREEP-PLAST
17. CRSSA
18. DAHRS
19. DEAP
20. DEBLIN2
21. DEMO
22. DOTIIIW or DOTIII
23. DRIPS
24. DUNHAM's
25. DYNALSS
26. DYNAPLAS
27. ELTEMP
28. ETOX
29. E0984A
30. E1682A
31. E1755A
32. E1739A
33. FATHOM-360 & 360S
34. FBRDSAP
35. FESAP
36. FLODISC
37. FORE-2M
38. FRST
39. FULMIX
40. FURFAN
41. GAMLEG-W
42. GASA
43. GASP
44. GSAP4
45. HAA-3B
46. HAFMAT
47. HAP
48. HAP-II (SAP)
49. HETHA/VETHA
50. HOTDAMG
51. HYTRAN
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56. LSD-2
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58. MARC
59. MENAC
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61. MRI/STARDYNE
62. NAPALM
63. NASTRAN
64. NICER
65. NONSAP
66. OCTOPUS
67. ORIGEN
68. PDA
69. PERT-V
70. PIP
71. PLAP
72. PUMA
73. QAD
74. RIBD-II
75. S-4
76. SAP-IV
77. SAP4GE
78. SCAP
79. SCAP-BR
80. SETS
81. SNAP
82. SOFIRE-II
82.A. SPCA
83. SPECEQ/SPEC
84. SPHINX
85. SPRAY-I
86. SUPERPIPE
87. TAP-A
88. TAP-B
89. TAP-4F
89A. TEMPEST
90. TFEATS
91. TGRV
92. THI-3D
93. THTD/THTE
94. TRANSWRAP
95. TRIPLET-W
96. TRITON
97. TRUMP
98. VARR-II
99. VENTURE
100. WECAN
101. WESDYN
102. W-2DB
103. WRAPUP D
104. XSRES-WIDX
105. MPHI
106. (RESERVED)
107. 772
108. 781
109. 907
110. 1017
111. 1027
112. 1036
113. 1374
114. 1392
115. 1671
116. 1691
117. 1823

118. BOSOR4
119. CODES
120. EQUILIN
121. FLUSH
122. MODPROP
123. RESPECTPLOT
124. STRUDL
125. THAVSA
126. 2DGENFRAME

A.1 AID

AID (Accident Inhalation Dose) is a computer program written primarily for the parametric analysis of the control room inhalation dose following a major reactor accident. AID can calculate thyroid, whole-body, bone and lung doses under various reactor containment and control building ventilation system conditions. Based on input data for meteorological conditions and ventilation system setups, the program first calculates: (1) the atmospheric diffusion factors as a function of distance and time after an accident; (2) the dilution factor inside the control room according to ventilation system parameters such as filtered air intake rate, recirculation rate, distance between containment building and control room, point of air-inlet and the filter efficiency. Subsequently, the external whole-body, internal whole-body, thyroid, bone and lung doses, are calculated by the program. The AID code is based upon the nuclear power plant control room ventilation and meterological models as described by Murphy and Campe (Reference 1). AID can also be used to calculate on and off-site radiation exposures for any other major releases of radioactive material.

Availability

The AID code is available at Burns and Roe in Oradell, New Jersey on an IBM 370/168 computer.

Verification

The AID code verification has been by hand calculations as recorded in Burns and Roe, Inc. internal, non-proprietary, documentation.

Application

AID is used to calculate the internal and external whole body, thyroid, bone and lung doses to control room operators following a major release of radioactivity to ensure compliance with Design Criterion 17.

References:

(1) K. G. Murphy and R. M. Campe, 13th AEC Air Cleaning Conference "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" August 1974).

(2) TY Byoun and J. N. Conway, The Proceedings of the 14th ERDA Air Cleaning Conference, "Evaluation of Control Room Radiation Exposure," CONF-760822, August 1976.

A.2 ANISN (ANISN-W) (Westinghouse Proprietary)

ANISN-W is an ARD version (Westinghouse Proprietary) of the ANISN code. The ANISN-W code solves the one-dimensional, energy dependent, linear Boltzmann transport equation with general anisotropic scattering for slab, cylindrical, and spherical geometries. ANISN-W solves forward or adjoint, homogeneous or inhomogeneous problems. The inhomogeneous problems may have a fixed volume distributed source or a specified angular dependent shell source at any mesh interval; fissions may be included for a subcritical system. Vacuum, reflective, periodic, white, or albedo boundary conditions may be specified. Time absorption calculations, concentrations searches, outer radius searches, buckling searches, or zone thickness searches are also solved. Cross sections may be input from a library tape and/or from cards. Fixed distributed sources or shell sources may be input from cards and/or from tape. The code also includes space point scaling or coarse mesh rebalance to accelerate the flux solution on inner iterations.

The discrete ordinates, or Carlson's S_n method, using diamond difference solution technique, is employed. The method is applicable to both neutron and gamma ray transport problems. The solution in the code will approach the exact solution of the Boltzmann equation with increasing orders of approximation as the space, angle, and energy mesh approaches differential size.

Availability

The ANISN-W computer code is available on the Westinghouse Power Systems CDC-7600 computers located atthe Monroeville Nuclear Center. The version in use was released from Westinghouse Astronuclear Laboratory (WANL) in August, 1970, and has been updated at ARD to satisfy CRBRP shield design analysis requirements.

Verification

Selective verification procedures have been used for the ANISN-W code solution. These procedures include comparisons with similar one-dimensional codes, two-dimensional codes, and experimental results. (Experiments have been performed at both Oak Ridge National Laboratory (ORNL) and Argonne National Laboratory (ANL). In addition a series of test cases is maintained on tape and is run periodically to verify consistency of results. Hand calculations are done when applicable to verify minor updates required for CRBRP shielding analysis.

The flux weighted neutron cross sections generated with the aid of ANISN-W are used as part of the methods to analyze the plate-type critical experiments. The accuracy of these reference design methods as applied to appropriate critical experiment measurements is documented in subsection 4.3.2 of the PSAR. Further verification is planned based on the ZPPR results for CRBRP Engineering Mockup Critical experiments.

Application

CRBRP applications of the ANISN-W code include prediction of: 1) shielding worth of shielding materials; 2) experimentally measured values of neutron and gamma flux or dose, 3) neutron and gamma flux environment at specific components, and 4) effect of design parameters on shield performance or environment. The applications are primarily in areas where conceptual design parameters are required or where experimental configurations are amenable to one-dimensional (slab, cylinder, or sphere) analysis. ANISN-W is used to test nuclear data and models prior to analysis by two-dimensional methods. Multigroup approximations and particle scattering approximations are tested in ANISN-W prior to two-dimensional analyses.

The Zero Power Plutonium Reactor (ZPR) series of critical experiments performed in support of the LMFBR program in general and in support of CRBRP in particular, are plate-type experiments in which fuel, structural and coolant materials are mocked-up in platelet geometry. The spacial cell homogenizations for the analysis of these critical experiments were performed using the ANISN-W code specially modified to perform the required cell flux weighting. Periodic boundary conditions were used for the plate cell calculations with plate independent values of buckling for cell leakage representation.

References

R. G. Soltesz and R. K. Disney, "Nuclear Rocket Shielding Methods, Modification, Updating and Input Data Preparation, Volume 4: One-Dimensional, Discrete Ordinates Transport Technique," WANL-PR-(LL)-034. August, 1970.

W. W. Engle, Jr., "A Users Manual for ANISN: A One Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," Union Carbide Corporation Report K-1693 (1967). (This document does not include the Westinghouse Proprietary Modifications.)

A.3 ANSYS (Proprietary - Swanson Analysis System)

ANSYS is an engineering analysis system which provides a flexible framework for implementation of the finite element analysis technology. Included in the ANSYS program are capabilities for static and dynamic, elastic and plastic, fluid flow, and transient heat transfer analyses. Based upon finite element idealization, the available library of finite elements numbers more than twenty for static and dynamic analyses and more than five for heat transfer and fluid flow analyses. These elements include plane stress and axisymmetric triangles, three dimensional solids, springs, masses, dampers, plates, axisymmetrical shells, general shells, friction interface and frame structure elements.

Availability

The ANSYS computer program is widely available in the United States and at specific locations in Canada, Europe, Australia, Africa, and Japan. The program can be used on a computer time-sharing basis, a royalty basis, or, in certain controlled circumstances, it may be leased or purchased from Swanson Analysis Systems, Inc., of Elizabeth, PA. It is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center, and on the Honeywell 6000 facilities of the General Electric Company, Nuclear Energy Systems Division.

Verification

The ANSYS program is in a state of continuous development. The verification of ANSYS is being undertaken by Swanson Analysis Systems, Inc. The ANSYS program has been used for production analyses since early 1970, and the user group now includes the nuclear, mining, chemical, electronic and automotive industries, and many consulting firms.

ANSYS is recognized and widely used in industry with a sufficient history of successful use to justify its validity per SRP 3.9.1, Section II.2.a.

Application

The ANSYS program has the capability of analyzing frame structures (two dimensional frames, grids and three dimensional frames), piping systems, two dimensional plane and axisymmetric solids, flat plates, three dimensional solids, axisymmetric and three dimensional shells and non-linear problems including interfaces and cables. ANSYS is applied to thermal and structural analysis of Reactor and Heat Transport System structures and components.

References

ANSYS Engineering Analysis System User's Manual, 1975, by G. S. DeSalvo, Ph.D. and J. A. Swanson, Ph.D., Swanson Analysis Systems, Inc., 870 Pine View Drive, Elizabeth, PA., 15037.

ANSYS Engineering Analysis System Examples Manual, 1975, by G. S. DeSalvo, Ph.D. and J. A. Swanson, Ph.D., Swanson Analysis Systems, Inc., 870 Pine View Drive, Elizabeth, PA., 15037.

"ANSYS Engineering Analysis System Verification Manual"; May, 1976. Swanson Analysis Systems, Inc.,870 Pine View Drive, Elizabeth, Pa. 15037.

A.4 APPROPOS

The APPROPOS code prepares multigroup cross section data in the format required for discrete ordinate transport codes and multigroup coupled neutron-gamma cross section data and neutron reaction cross section data for radiation transport analysis.

The APPROPOS code processes multigroup, neutron cross section data and multigroup, neutron reaction cross section data (e.g., radiative capture, fission, inelastic scatter, and elastic scatter) to provide spectral-weighted, neutron cross section data in a reduced number of energy groups. These data are then processed with input specified elemental atom densities, gamma ray production data due to neutron reactions and gamma cross section data to provide macroscopic, coupled, neutron-photon cross section data, neutron cross section data, and gamma cross section data. Output data from APPROPOS is used in the ANISN-W or DOTIIIW codes.

Availability

The APPROPOS code is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center. The version currently being used was released from WANL in August, 1970, and has been updated at ARD to satisfy CRBRP shield design analysis requirements.

Verification

The cross section preparation and coupling technique in APPROPOS have been verified by hand calculations. The cross section preparation and coupling technique in APPROPOS, as utilized in CRBRP anlayses, is a data processing function to reformat basic nuclear data into the formats required in radiation transport methods.

Application

APPROPOS is used to process multigroup neutron cross section data, neutron multigroup reaction rate data, and gamma ray production data into spectrally weighted, group collapsed, coupled macroscopic cross sections for use by the ANISN-W, DOTIIIW, TRIPLET-W, and the many editing codes used in CRBRP shielding analysis.

Reference

R. G. Soltesz, R. K. Disney, and S. L. Zeigler, "Nuclear Rocket Shielding Methods, Modifications, Updating, and Input Data Preparation, Volume 3: Cross Section Generation and Data Processing Techniques," WANL-PR-(LL)-034, August, 1970.

A.5 APSA Finite Element Axisymmetric and Planar Structural Analysis with Orthotropic Temperature Dependent Material Properties.

The APSA Code applies the finite element method to the determination of displacements, stresses, and strains in axisymmetric and planar solids. Two classes of material behavior, isotropic, and orthotropic linear elasticity, can be modeled. The material properties can beinput as a multilinear function of temperature. During solution of a problem, each element temperature is used to obtain element material properties from the tabular input by linear interpolation. The mechanical loads can be surface pressures, surface shears, and nodal point forces as well as axial acceleration and angular velocity. The thermal load is input as a temperature field at specified nodes. The element used in the APSA Code is a modified Iron's quadrilateral. In addition to the printed output, a data tape is generated containing the input and the calculated results. A CRT plotting program uses the tape to generate a variety of plots such as geometry mesh, stress contours, etc.

Availability

APSA, as released in April 1974, is available on the IBM 370 computer of Rockwell International Western Computing Center.

Verification

Documentation of verification of the APSA Code, per SRP Section 3.9.1.II.2.c, can be found in the reference.

Application

The APSA Code was used in the analysis of CRBRP steam generator.

Reference

Newell, J.F.; Persselin, S.F., "Finite Element Axisymmetric and Planar Structural Analysis," Report No. SSME 74-1282, Rocketdyne, Canoga Park, California 91304, April 1974.

A.6 ASHSD2

ASHSD2 is a computer program to evaluate the time dependent displacements and stresses of complex axisymmetric structures subjected to any arbitrary static or dynamic loading or base acceleration. The three dimensional axisymmetric continuum is represented either as an axisymmetric thin shell or as a solid of revolution, or as a combination of both. The axisymmetric shell is discretized as a series of frustrums of cones and the solid of revolution as triangular or quadrilateral toroids connected at their nodal point circles.

The program can solve five cases of loading: dead load, arbitrary static load, arbitrary dynamic load, horizontal and vertical component of earthquake acceleration record applied at the base of the finite element model.

Any arbitrary loading is first approximated by a Fourier series with a finite number of terms. For each Fourier component the stiffness and mass matrices and the corresponding load vector are formed and the equation of motion is solved through the time domain either by direct integration or by mode superposition using a numerical step-by-step integration procedure. After solving for the response of all the Fourier terms their contributions are summed up to obtain the total response.

ASHSD2 is written in FORTRAN IV for CDC 6000 series computers. The program uses dynamic storage allocation. A separate version of the program for static analysis only (ASHSAB) contains in-core as well as out-of-core equation solvers. Program ASHSAB is included in the complete program package.

Availability

ASHSD2 is available on the CDC 7600 computer of Lawrence Berkeley Laboratory. The current version was released on September, 1975.

Verification & Comparison

Verification of ASHSD2 per SRP Section 3.9.1.II.2C is documented in the Reference using sample problems discussed therein. A comparison against an analytical solution (taken from the Reference) is given in Figure A-1.

Application

ASHSD2 will be used in the stress analysis of the sodium dump tanks and the reaction products separator tanks.

Reference

Gosh, S. and Wilson, E. L, "Dynamic Stress Analysis of Axisymmetric Structures under Arbitrary Loading," University of California, Berkeley, September 1969 (EERC 69-10). (Revised September 1975 by C. J. Lin.)

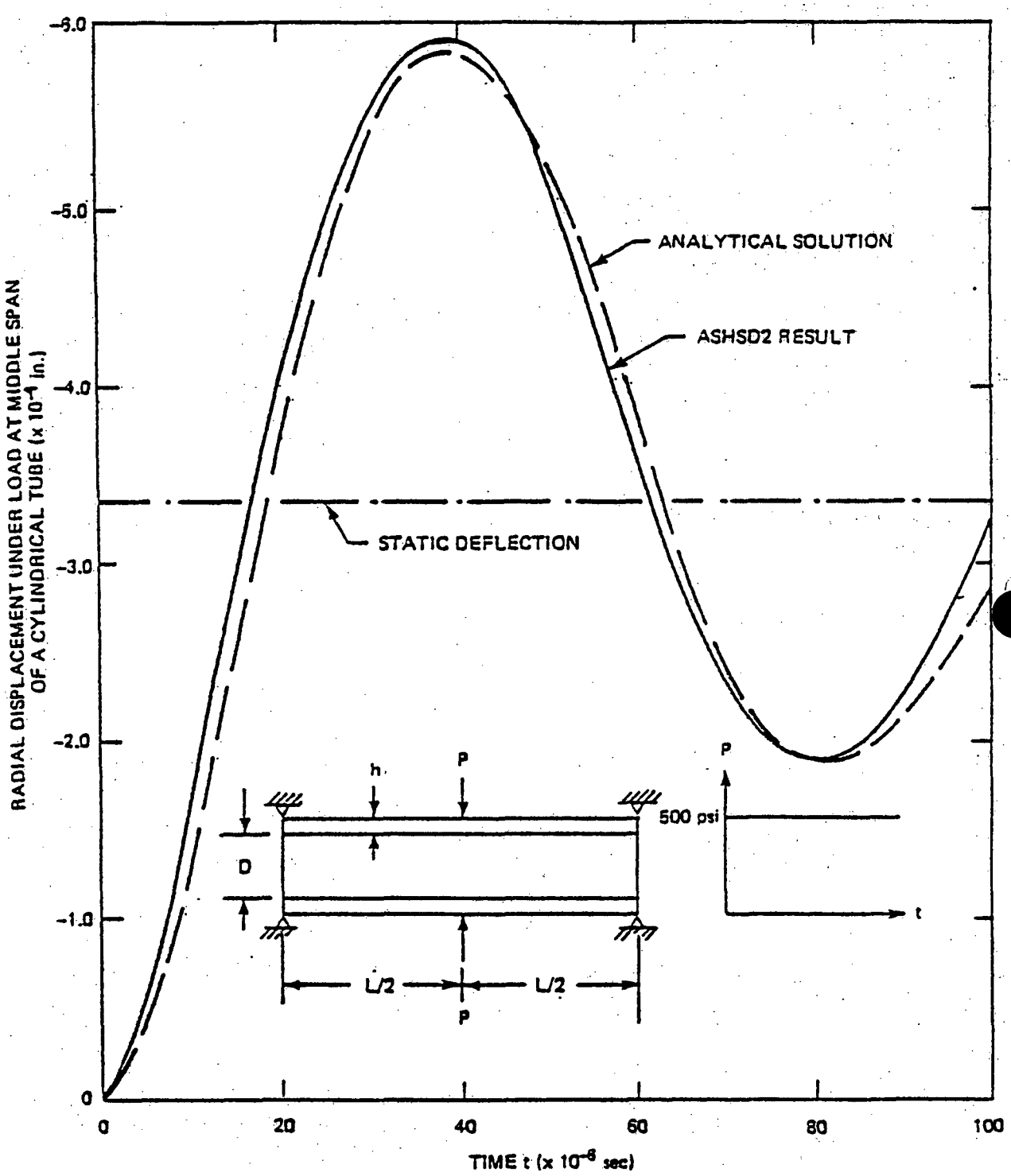

Figure A.6-1 Comparison of ASHSD2 Against Analytical Solution

A.7 Autotem-III (Westinghouse Proprietary)

Autotem-III consists of nine independent programs written in the FORTRAN language. The first six must be executed together to generate a nodal network from input geometric specifications. The seventh program combines the nodal network with input boundary conditions to generate a complete TAP-A input file. The eighth program is the TAP-A code, which calculates temperature distributions. The last program plots isotherms on the nodal pattern.

Availability

Autotem-III (July, 1975 version) is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

Verification consists of checking the physical dimensions of meshes generated for arithmetic correctness, and the comparison of calculated results against closed-form solutions and calculated results from other verified computer codes. It is planned that verification will be completed by August, 1978.

Application

Autotem-III is used to generate nodal models for, and solve, heat transfer problems in general physical systems.

References

To be provided.

A.8 CACECO

The CACECO code computes pressure, temperature, and mass constituent transients in four connected cells by performing energy and mass balances. The cells are composed of a series of one-dimensional thermal structures composed of several materials. One material may be concrete which liberates water vapor and carbon dioxide as it is heated. Each cell may have a molten pool consisting of water or sodium and an atmosphere consisting of water vapor, sodium vapor, nitrogen, carbon dioxide, oxygen and/or hydrogen. Chemical reactions considered in the cells include sodium-concrete, sodium-water, sodium-carbon dioxide, sodium-hydrogen, hydrogen-oxygen, and sodium oxide-water. Venting and purging of the cells may be simulated. Energy input to the cells can also be considered. CACECO was developed by the Hanford Engineering Development Laboratory (HEDL).

Availability

The CACECO code is available from the Argonne National Laboratory Code Center. It is operational at HEDL, GE Breeder Reactor Devision, Westinghouse Advanced Reactors Division, Brookhaven National Laboratory, Nuclear Regulatory Commission, and University of California at Berkley.

Verification

The CACECO code was used extensively in the evaluation of post accident transients for the FFTF reactor (HEDL-TC-222, November 1974 and HEDL-TME-77-18, April 1977). In the application to FFTF, the code was checked many times at HEDL and was found to execute the equations properly.

In the CRBRP application, many of the subroutines have been verified by comparison with hand calculations and other computer programs. For example, the conduction subroutine compares very well with a standard heat conduction code (TRUMP) as shown on Figure A.8-1.

Further verification is planned, including both analytical checks and comparison of CACECO results with experimental data.

Application

CACECO can be used for analysis of containment cells following sodium spills or hypothetical core melts. It is currently used in studies of Thermal Margin Beyond the Design Base and in sodium spill analyses.

Reference

R. D. Peak, "Users' Guide to CACECO Containment Analysis Code," HEDL-TC-859, June, 1977. (Availability-USDOE Technical Information Center).

COMPARISON OF TEMPERATURE DISTRIBUTIONS
THROUGH THE REACTOR CAVITY WALL AT 80 HOURS

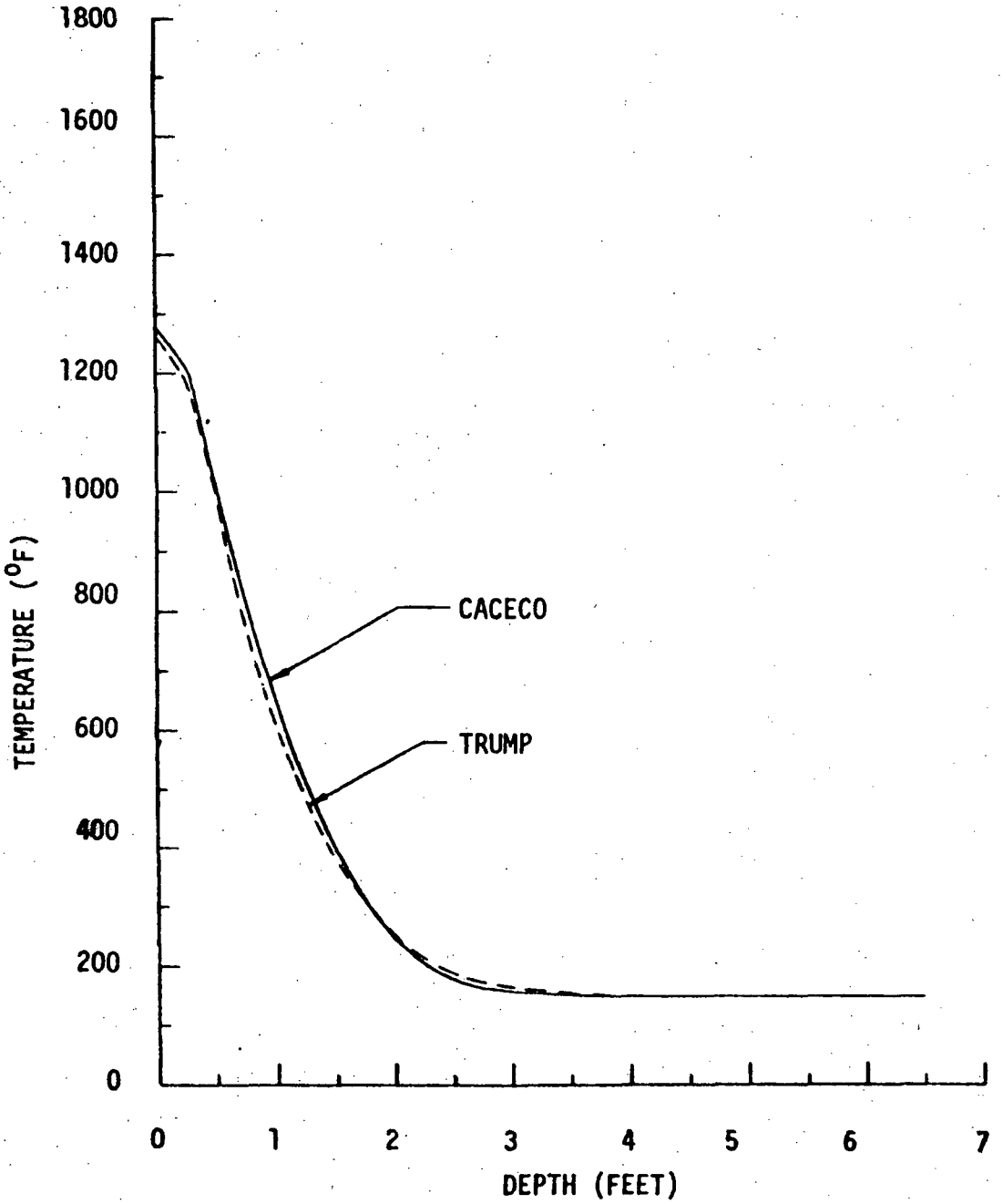

FIGURE A.8-1

A.8A CATFISH

The CATFISH code analyzes the hydraulic network of the primary system of a nuclear reactor employing ducted assemblies. The code models the flow resistance in the core, the inlet and outlet plena and the primary loop. A branched network represents the numerous flow paths in the reactor: fuel and blanket orificing zones, primary and secondary control assemblies, radial shield, vessel and core barrel and leakage. For each individual core assembly, represented by a separate flow path, every hydraulic resistance is accounted for: lower internals, assembly nozzles, orifice, shield, rod bundle, etc. The entire network is coupled with the pump head/flow characteristics curve. Thus, for any specified set of resistances, CATFISH calculates the pump head, the total reactor flow, the flow in each assembly and the pressure drop across each subcomponent. Also (this is the case for CRBRP design), for a specified core flow distribution obtained through ØCTØPUS, CATFISH calculates the orifice hydraulic resistances in each orificing zone and the total reactor flow compatible with the pump specification. Additionally, it calculates the pressure drop in each component of the primary system and the individual flow in each assembly (assemblies with identical orificing, but fed by different lower inlet modules (LIMs) will have slightly different flows, depending on the pressure drop across the respective LIMs, which is proportional to the total LIM flow).

CATFISH calculates core flow distribution and components pressure drops for nominal conditions as well as accounting for uncertainties on the hydraulic resistances, both positive and negative (higher or lower resistance than nominal). The code also has the capability to calculate statistical variation in total reactor flow by individually varying each hydraulic resistance, calculating the corresponding reactor flow variation and the root mean square of the individual variations at a specified level of confidence. Uncertainties can be combined systematically or semi-statistically in any desired flow path(s) in addition to the fully statistical combination discussed above.

Availability

CATFISH is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

Hydraulic resistances and associated uncertainties used in CATFISH are obtained from prototypic testing of assemblies and reactor internals components for the FFTF and CRBRP reactors. These values are continuously being updated as new experimental data become available.

The method of solution adopted in CATFISH has been verified by hand calculations for a simplified network. More detailed verification is planned by comparing CATFISH predictions against HAFMAT and CØRINTH.

Application

The CATFISH code is used in conjunction with the ØCTØPUS code to calculate the flow in each individual fuel and blanket assembly. Additionally, CATFISH is used to calculate the flow rate in other core assemblies, the pressure drop in each primary system component and the orificing requirements
53 in the core assemblies.

References

1. M. D. Carelli and J. M. Willis, "An Analytical Method to Accurately Predict LMFBR Core Flow Distribution", paper presented at ANS 1979 Summer Meeting, June 1979.

2. M. D. Carelli and J. M. Willis, "Analytical Modeling of Core Hydraulics and Flow Management in Breeder Reactors", paper submitted to XVIIIth International Association for Hydraulic Research (IAHR) Congress, Cagliari (Italy), September 1979.

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A.9 CHERN

The CHERN code provides an analytical solution to the elastic-plastic-creep problem of a long thick-walled cylinder under cyclic loading conditions. Time dependent internal/external pressure, thermal gradients, as well as axial tractions or displacements can be applied in a non-proportional manner. The solution incorporates the latest ORNL recommended constitutive relationships which will be incorporated into RDT Standard F-9. The program is a substantial improvement over Bree's technique and is used as a design aide for components operating in the creep range.

Availability

The CHERN version described in Ref. 1 has been available on the CDC 7600 computer of Lawrence Berkeley Laboratory since March, 1975, and on the Westinghouse Power Systems CDC 7600 computers at the Monroeville Nuclear Center.

Verification

CHEN has been compared with the most widely used inelastic computer codes (MARC, ANSYS and CREEP-PLAST; Ref. 1 and 2). A summary comparison is provided in Figure A.9-1.

Application

It will be utilized for inelastic analysis of long cylindrical components. Applications will be to demonstrate satisfaction of the deformation controlled limits of ASME Code Case 1592.

References

(1) J. M. Chern, "Elastic-Plastic-Creep Analysis of a Thick-Walled Cylinder Subjected to Time Dependent Pressure, Temperature and Axial Loads or Displacements", FWR-35, Rev. 0. Foster Wheeler Corp., Livingston, N. J., July 1972.

(2) "Intermediate Heat Exchanger for Fast Flux Test Facility Evaluation of Inelastic Computer Programs," Prepared by Foster Wheeler Vorporation for Westinghouse Advanced Reactor Division, FWR-27, Rev. 0, 3/2/72.

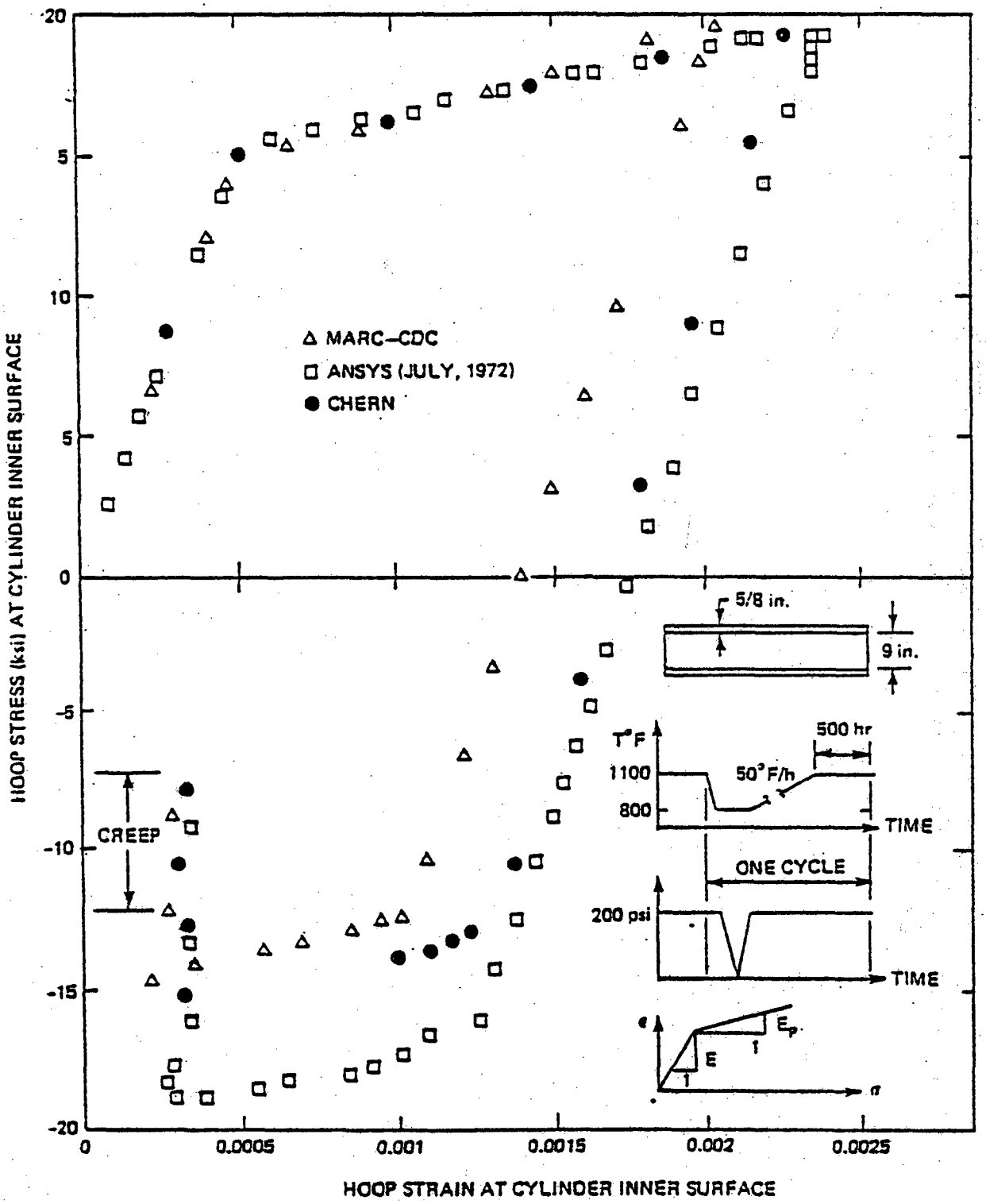

Figure A.9-1. Comparison of CHERN with the MARC and ANSYS Computer Codes

A.10 CINDA-3G

The Chrysler Improved Numerical Differencing Analyzer (CINDA) program offers a variety of methods for solution of thermal analog models presented to it in a network format. Numerous subroutines are included to handle diffuse radiation in enclosures, fluid transport, sublimation, etc.

Availability

CINDA-3G is available from the Chrysler Corporation Space Division, New Orleans, La. Report No. TN-AP-67-887. It is used on the CRBRP project by the Foster Wheeler Energy Corporation.

Verification

The results of the CINDA-3G program have been checked against those of the well verified thermal computerprogram "THTD" (A.93), for the case of a simplified heat transfermodel. The results from both computer codes are identical which verifies CINDA-3G under acceptance criteria II.2.6 of SRP 3.9.1.

Application

CINDA-3G is used for steady-state thermal modeling at full and part load and dry heat-up analysis for the CRBRP-IHX and pre-heat analysis of the CRBRP PHTS check valve.

Reference

Report No. TN-AP-67-887, Chrysler Corporation Space Division, New Orleans, La.

A.11 CØBRA

The CØBRA type codes are thermal-hydraulic subchannel analysis codes of the "rigorous" type, i.e., the flow and temperature distributions inside an assembly during steady state and transient conditions, either of the wire wrapped or gridded type, is calculated by solving simultaneously the conservation equations of mass, momentum and energy. It uses a mathematical model that considers both turbulent and diversion crossflow mixing between adjacent subchannels. Each subchannel is assumed to contain one-dimensional, two-phase (for LMFBR accident analysis applications), separated, slip-flow. The two-phase flow structure is assumed to be fine enough to define the void fraction as a function of enthalpy, flow rate, heat flux, pressure, position and time. At the present time, steady state two-phase flow correlations are assumed to apply to transients. The mathematical model neglects sonic velocity propagation; therefore, it is limited to transients where the transient times are greater than the time for a sonic wave to pass through the channel. The equations of the mathematical model are solved by using a semi-explicit finite difference scheme. The scheme also gives a boundary-value flow solution for both steady state and transients where the boundary conditions are the inlet enthalpy, inlet mass velocity, and exit pressure. A program is underway to expand and modify the CØBRA capability to predict subchannel performance during undercooling transients, accounting for the effects of inter- and intra-assembly flow and heat redistribution on a core-wide basis. This is referred to as the "whole core" version of CØBRA.

Availability

The CØBRA codes are available through the Argonne National Laboratory Code Center at Argonne, Illinois or through the Battelle Memorial Institute's Pacific Northwest Laboratory at Richland, Washington. The program is used at Westinghouse on the Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

The verification of CØBRA for LMFBR analyses is being performed by BPNL by comparison to test data from various sources relevant to heat transfer and flow distribution in rod bundles and reactors. These sources include the HEDL 217-pin low flow heat transfer test, ARD 61-pin blanket heat transfer test, ORNL 19-pin FFM and 61-pin THORS tests, EBR-II and FFTF steady state and transient tests.

Application

The CØBRA codes are used to solve the bulk coolant flow and temperature distribution for rod bundle nuclear fuel element subchannels of fuel and radial blanket assemblies. The code is also used in analyzing the hot spot factor corresponding to variations of the subchannel flow area caused by geometrical tolerances or bowing. In addition, the code is used to model heat transfer through the inner duct of the primary control assemblies, i.e., between the absorber bundle and flow bypass. Thus, the coolant flow and temperature distribution inside the primary control assembly absorber bundle may be determined.

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References

1. C. W. Wheeler, et al., "CØBRA-IV-I: An Interim Version of CØBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and
53 Cores", BNWL-1962, March 1976.

A.12 COMRADEX III

This program was developed by Atomics International to estimate potential radiological consequences form postulated reactor accidents considering the effect of containment and meteorology on the resultant environmental radiation exposure. Reference 1 provides a description of the COMRADEX Code.

Availability

COMRADEX is currently available through the Argonne National Laboratories Code Center and on the Westinghouse Power Systems CDC 7600 computers located at the Monroeville Nuclear Center.

Verification

The validation of COMRADEX by comparison with hand calculations is documented in Reference 2.

Application

COMRADEX-III is used for the calculation of radiological doses resultant from postulated nuclear power reactor accidents.

References

1. J. M. Otter and P. A. Conners, "Description of the COMRADEX-III Code," TI-001-130-053, Atomics International, June 30, 1975.

2. P. A. Conners, R. S. Hart, and J. M. Otter, "COMRADEX III Code Validation," N099TI120005, Atomics International, August 22, 1977.

A.13 CONLIFE

CONLIFE Code is a modified version of the CONROD code, (Reference) which was written for the FFTF project to compute the absorber pin lifetime. The principal modifications to the CONROD code have been the elimination of the assembly hydraulics so the code now handles a single absorber pin, updating of the material properties, incorporation of the CRBRP heating rates, and updating of the stress calculations. The code computes temperatures, gas generation and release, boron carbide swelling, cladding swelling, and cladding stress and strain. The code can handle fixed or moving control rods, and performs thermal calculations for average, peak, and hot channel pins.

Availability

CONLIFE has been available on the Honeywell 6000 computer system of General Electric in San Jose.

Verification

The CONROD code has been extensively compared to other computer codes and hand calculations.

Application

The CONLIFE code will be used to compute the bahavior of the SCRS absorber pins.

Reference

K. R. Birney and A. L. Pitner, "User's Manual for CONROD - A Computer Program for the Design Analysis of LMFBR Control Elements," HEDL-TME 75-131, December, 1975.

A.13A CØRINTH

CØRINTH is an extensively modified version of the transient FLØDISC code (Ref. 1). The modifications provide the capability to predict the inter-assembly flow redistribution during undercooling transients, considering the unique characteristics of LMFBRs. These characteristics include the branched hydraulic network introduced by the lower inlet modules of CRBRP, the different transient power decay rates in fuel, blanket and absorber assemblies and the experimentally determined hydraulic characteristics and rod bundle friction factors of the different assembly types (see Figures A-36-1 and A-36-2). Additional capabilities include time varying reactor inlet temperature, axial heat conduction in the coolant, flow rate-dependent film coefficients and the prediction of either forward or reversed flow. The CØRINTH model is based on one-dimensional coolant flow (no intra-assembly heat or flow redistribution) with a lumped parameter transient model and no inter-assembly heat transfer.

Availability

The CØRINTH code is being developed by W-ARD and is operating on the Westinghouse Power Systems CDC-7600 computers located at the Westinghouse Monroeville Nuclear Center.

Verification

Verification of the steady state capability of CØRINTH will be the same as for FLØDISC. The transient capability will be verified by comparison to experimental data and predictions from similar transient codes. The experimental data will include comparison to results from the FFTF Acceptance Test Phase (ATP) tests.

Application

CØRINTH will be used in future PSAR revisions to predict inter-assembly flow redistribution during natural circulation transients. Such predictions will replace the calculations currently performed with the steady state FLØDISC code.

References

1. J. Muraoka, "FLØDISC: A Dynamic Core Flow Distribution Code - Evaluation of the Total Loss of Electrical Power Event", HEDL-TC-874, May 1977.

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A.13B CONROD VERSION II

An absorber pin lifetime behavior prediction code for analysis of breeder reactor control assembly bundles. Options provide applicability to general geometries and a variety of material types and operating conditions. Parallel analysis and multicase structure allow fast comparison between different designs and materials properties correlations. In conjunction with uncertainty factors on temperatures, power, and absorber material behavior, worst condition design cases are easily identified. The code is written in FORTRAN IV and uses free field input format.

AVAILABILITY:

Currently maintained on the Lawrence Berkeley Laboratory 6600/7600 computer system as a part of controlled library on the Program Storage System. The code last developed on the 7600 operating system BKY30CV7. The code is controlled and maintained by the Fuel Design unit of ARSD.

VERIFICATION:

The original versions of the code were written by HEDL specifically for FFTF. The background theory, and the engineering review of both code and typical analysis is documented in Ref. 1. The internal review of the code at HEDL during development, and the fact that it has been the primary absorber pin design tool for FFTF is construed as a partial verification.

To apply the code to the SCA, modifications were necessary to generalize the modeling capabilities and design philosophies. These modifications and the line by the comparison of the code with the published theoretical equations of Reference 1 which accompanied the modification, are being documented in Reference 2. The items remaining which will complete the code verification are:

a) Develop benchmark solutions for the thermal, hydraulic, and structural methods of the code.

b) Evaluate the code predictions against these benchmark solutions.

c) Document the Evaluation (in Ref. 2).

APPLICATION:

The code has been used to analyze absorber pin bundle designs for the CRBRP Secondary Control Assemblies and for the FFTF. The nucleonics data, and thermal hydraulic boundary conditions are those usually available from core wide analyses.

REFERENCES:

1) Pitner, A. L., "User's Manual for CNRD2. A Design Analysis Code for LMFBR Control Rods"; HEDL-TMI-81-50, Hanford Engineering Development Laboratories, Richland, WA.

2) Heisser, D. J., "User's Guide for CONROD II" (to be published).

A.13C CORTEM

The CORTEM computer code analyzes the steady state thermal-hydraulic behavior of a 30-degree sector of an LMFBR core, consisting of fuel, blanket and control assemblies. CORTEM consists of an unique modeling of three-way flow split including downflow in Secondary Control Assembly in CRBRP. The code treats steady state intra-assembly and interassembly heat transfer in a full 30-degree sector of the core. The intra-assembly heat transfer inside core assemblies is modeled based on application of the subchannel concept together with the use of bulk parameters for coolant velocity and coolant temperature within a subchannel. The interassembly heat transfer between assemblies is determined by heat transfer coefficient in the assembly gaps which is a function of interstitial flow and Peclet number of the coolant.

AVAILABILITY:

The CORTEM code is available on GE Honeywell 6000 computer located at GE/NEBO, San Jose, and CDC7600 at Berkeley.

VERIFICATION:

Results from the CORTEM code have been compared with results from other computer codes. Good agreement was found and is documented in the reference.

APPLICATION:

The CORTEM code is used to predict the steady-state thermal-hydraulic behavior of an LMFBR core, including the effects of intra-assembly and interassembly heat transfer. In particular, it has been used to set the correct thermal (heat transfer) boundary conditions for the CRBRP Secondary Control Assemblies.

REFERENCE:

J. P. Wei, "A Simplified Interassembly Heat Transfer Model for the Analysis of Liquid-Metal Fast Breeder Reactor Core Restraint Systems", Nuclear Technology, Vol. 46, No. 1, November 1979.

A.14 CØTEC

CØTEC is a phenomenological subchannel analysis code used to investigate the flow and temperature distributions in wire wrapped assemblies. The code accounts for the following four mechanisms responsible for inter-channel heat and mass transfer: a) pumping due to change in subchannel area and hydraulic diameter caused by wire rotation; b) sweeping caused by fluid following the angle of the wire under the projection of the wire wrap; c) mixing due to turbulent heat transfer; and d) thermal conduction. The CØTEC code solves the continuity and energy conservation equations, and utilizes empirical fit of variable input parameters rather than a rigorous solution of the momentum conservation equation.

Availability

CØTEC is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

Empirical factors in the CØTEC code to model the effects of turbulent mixing, pumping (forced flow following the wire wrap), sweeping in the interior channels and swirl in the edge channel, are selected by calibration of the code against available experimental data. A comparison and calibration of the CØTEC code with available experimental data from ORNL (19-rod bundle FFM tests), ANL (91-rod bundle mixing tests), HEDL (217-rod bundle mixing tests), WRL (11:1 scale section of a 217-pin wire wrapped rod bundle air flow test) and JOYO (experimental study on coolant mixing effect in JOYO 19-rod bundle blanket assembly) was performed.

Additional comparison and calibration of CØTEC is being performed against experimental data which recently became available, i.e., the MIT 61-rod salt injection in water, the KFK 61-rod heating in sodium and the JOYO 91-rod heating in sodium.

The code will be further calibrated against experimental data as they become available from the Westinghouse ARD blanket heat transfer test, ORNL 61-rod fuel assembly bundle test and WRL air flow tests of a 5:1 scale section of a 61-pin blanket assembly.

CØTEC predictions have been, and will continue to be, compared against those of other subchannel analysis codes, chiefly the benchmark codes CØBRA and THI-3D.

Application

The CØTEC code is used to solve the nominal flow and temperature distribution of wire wrapped hexagonal rod bundles where the flow rate is above
53 the laminar regime and the effects of buoyancy can be neglected.

References

1. E. H. Novendstern, "Mixing Model for Wire Wrap Fuel Assemblies", Trans. American Nuclear Society, 15, pp. 866-867 (1972).

2. E. H. Novendstern, "Turbulent Flow Pressure Drop Model for Fuel Rod Assemblies Utilizing a Helical Wire Wrap Spacer System", Nucl. Eng. and Design, 22, pp. 19-27 (1972).

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A.15 CRAB (Westinghouse Proprietary)

The CRAB computer code analyses the steady state hydraulic and scram dynamics behavior of control assemblies of the poison rod bundle type. The code models the complex control assemblies flow network, and the flow distribution inside the wire-wrapped absorber bundle and in the bypass is calculated by satisfying the boundary condition of equal pressure drop across the individual channels.

The transient portion of the CRAB code features a complete evaluation of scram dynamics characteristics (control bundle insertion, velocity, acceleration, pressure drop as function of time). The equation of motion for a control volume encompassing the entire rod bundle is solved simultaneously with the transient hydraulics conservation equations by means of a fourth-order Runge-Kutta double integration process. Effect of assembly weight and buoyancy, scram spring, bellows, sliding friction, fluid shear forces and time-dependent forces such as seismic retarding forces during an earthquake is analyzed by CRAB. Both nominal and misaligned conditions can be investigated.

Availability

The CRAB code is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

Very satisfactory agreement was found in comparing CRAB predictions with FFTF control assembly mockup experimental data, both for steady state hydraulics, Figure A.15-1 (Fig. 1 from Ref. 1 for summary) and scram dynamics characteristics, Figure A.15-2 (Figures 2 and 3 from Ref. 2 for summary). Further verification of the code against CRBRP experimental data is planned following completion of the CRBRP control assembly prototype scram and hydraulic testing.

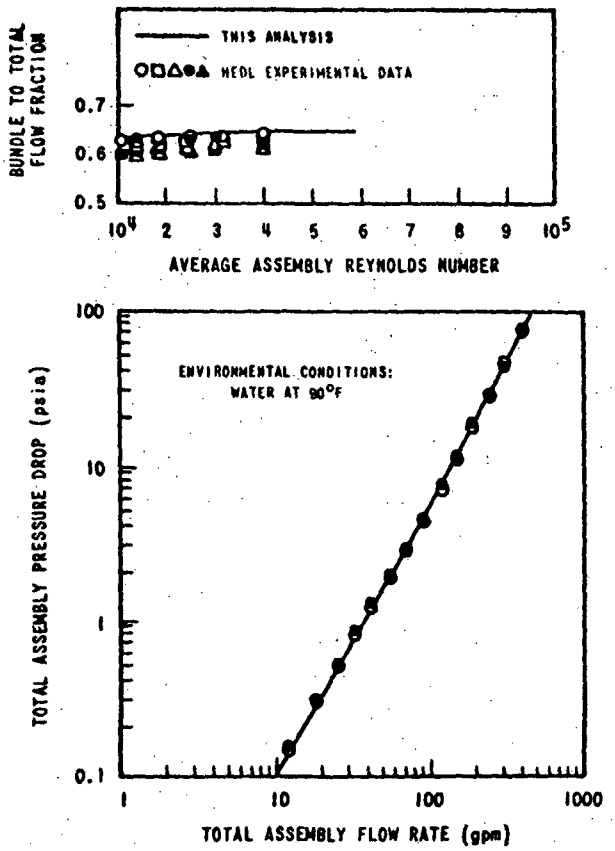
Application

The CRAB code is used to predict the steady state hydraulics and scram dynamics behavior for the T&H design of control elements of the rod bundle type.

References

1. M. D. Carelli, C. W. Bach, R. A. Markley, "Hydraulic and Scram Dynamics Analysis on LMFBR Control Rod Assemblies", Trans. Am. Nucl. Soc., 16, pp. 218-219 (1973).

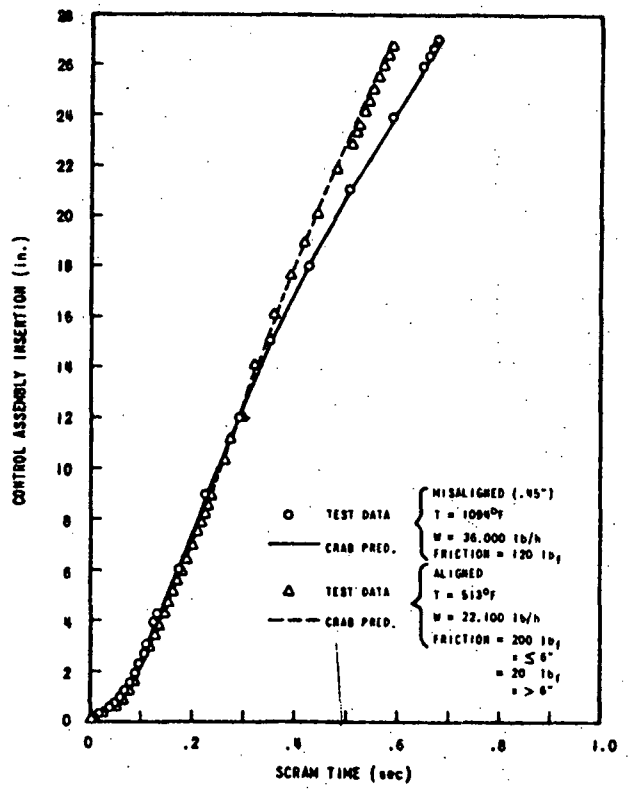
2. M. D. Carelli, C. W. Bach, H. W. Brandt, H. D. Kulikowski, "LMFBR Control Rod Scram Dynamics", Trans. Am. Nucl. Soc., 18, pp. 278-279 (1974).



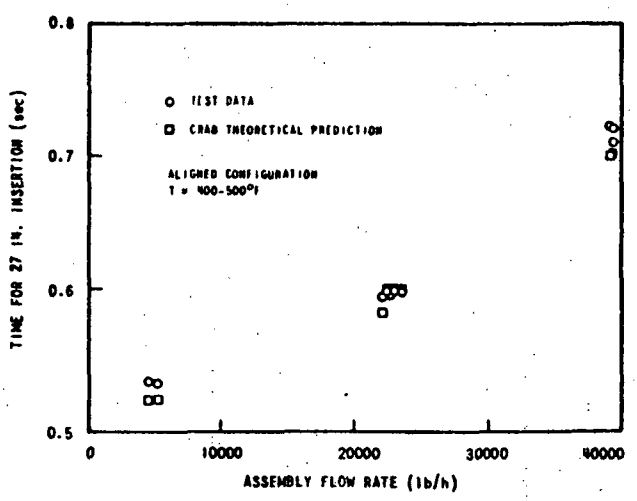
Comparison between theoretical predictions of this analysis and HEDL experimental data of the FFTF control assembly hydraulic behavior.

From Reference 1, CRAB code

Figure A.15-1



Comparison of predicted and observed scram insertion for FFTF control assembly models.



Comparison of predicted and observed insertion times for FFTF control assembly low-temperature tests.

From Reference 2, CRAB code

Figure A.15-2

A.16 CREEP-PLAST

CREEP-PLAST is a two-dimensional, non-linear, elastic-plastic-creep, finite element computer program for metal structures operating at high temperatures. The program considers instantaneous (time-dependent) elastic-plastic and time-dependent creep simultaneously in an incremental procedure which requires the user to select his own discrete time and/or load increments. The program is capable of predicting deformations, stresses, and total accumulated strains as functions of time and/or load for user specified thermal and mechanical loadings which can be varied independently.

Availability

CREEP-PLAST (March 1973) is available on the UNIVAC 1108 computer of Information Systems Design, Santa Clara, California.

Verification

It has been widely used in the nuclear industry for inelastic analysis. ORNL has independently verified the results predicted by CREEP-PLAST as compared with experimental programs (Ref.). A good correlation has been well established.

Application

It will be applied to two-dimensional plane stress, plane strain and axisymmetric problems. It has been used for inelastic analysis of the steam generator nozzle.

Reference:

Rashid, Y. R., "Part 1, Theory Report for CREEP-PLAST Computer Program: Analysis of Two-Dimensional Problems under Simultaneous Creep and Plasticity", GEAP-10546, General Electric Company, January, 1972.

A.17 CRSSA (Westinghouse Proprietary)

Thermal, hydraulic and mechanical analyses during the primary control assembly lifetime for various assembly core positions and axial control rod locations are performed by means of the CRSSA code. The hydraulic section of the code is almost identical to the steady state portion of the CRAB code calculating the bundle/bypass flow split and the control rod pressure drop. The thermal section of the code is an adaption of the NICER thermal model to the particular case of primary control assemblies; pellet, cladding and coolant temperature distribution over the absorber section of the control rod is calculated for average, peak or hot pin depending on the adopted nuclear and engineering hot channel factors. The above thermal calculations are performed over a time period corresponding to the assembly lifetime.

The nuclear input to the code is used to calculate absorber neutron captures, helium production, and heat generation for varying control rod configurations and positions. The irradiation induced swelling of the cladding is also calculated. The mechanical section of the code calculates the stresses and strains in the cladding due to the internal pressurization from the helium production at the operating temperatures. The resulting cladding stresses are compared to allowable stresses for the operating temperatures and fluences and the changes in pellet-to-cladding diametral clearance from cladding and pellet swelling are calculated to determine control assembly lifetime. Consideration of cumulative cladding damage function in evaluating lifetime is included.

Availability

The CRSSA code is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

Verification of steady state hydraulics performed for CRAB applies to CRSSA. Individual channel coolant temperatures and flows are input from subchannel analysis codes. Pin temperature calculations are straightforward application of Fourier one-dimensional heat transfer law employing experimentally derived correlations for materials thermal properties, as reported in the Reference.

The structural lifetime calculations are based on the same methods used for fuel rods. Therefore, verification of FURFAN will apply to the structural lifetime methodology in CRSSA. Pellet swelling and helium release calculations over the control assembly lifetime are based on B_4C correlations contained in the Reference.

Application

The CRSSA code is used to predict the lifetime behavior of the CRBRP primary control rods.

Reference

"A compiliation of Baron Carbide Design Support Data for LMFBR Control Elements", HELD-TME-75-19, March 1975 (Availability: USDOE Technical Information Center).

A.18 DAHRS

DAHRS, a heat transport code, is a quasi-steady state simulation in that core outlet and steam generator sodium outlet temperatures are computed from steady-state heat balance expressions for the power, flows, and temperatures at each time step. Lumped parameter differential equations are used throughout the simulation except for the reactor core and steam generators. Coolant nodes are treated as "perfect mixing" chambers of uniform temperature and respond to inlet temperature changes with a time constant based on the mass to flow ratio. The equations are differenced in time and arranged in explicit expressions of temperature as a function of time step duration and temperature at the previous time. Each equation in the simulation is examined for the allowable time step and the lowest value is selected by the code. As flow decreases, the time step may increase, enabling efficient computation of up to 5-10 hours of plant behavior.

Availability

DAHRS (December 1976) is available on the Honeywell 6000 computer of General Electric in San Jose.

Verification

The DAHRS code is a heat transport code which utilizes well accepted basic principles in its formulation. Because of this, validation against test data is not planned. Verification of the numerics through independent review is in progress.

Application

The DAHRS computer code models the CRBRP heat transport system for the purpose of evaluating Steam Generator Auxiliary Heat REmoval System (SGAHRS) transient heat loads. Decay heating and coolant flows are inputs to the code. Stored heats in metal and sodium are accounted for in all parts of the reactor and main cooling systems. To check code performance, a subroutine continually calculates a plant heat balance accounting for heat in, heat out, and changes in stored heat throughout the plant. Material properties and heat transfer coefficients are assumed to be constant.

A.19 DEAP

DEAP is a descendant of theTAP computer program. TAP's program logic was revised and the program capabilities enlarged. DEAP has the capability to solve any existing TAP problem with only minor changes to the data deck.

Availability

DEAP has been available at Rockwell International on the IBM 360 computer since 1969. It is a Rockwell International proprietary computer code.

Verification

The computer program has been widely used in U.S. Air Force and NASA programs. It is used and recognized in industry and has a sufficient history of successful applications to justify its validity. This constitutes verification per SRP Section 3.9.1.II.2.a.

Application

DEAP is being used to calculate transient and steady-state temperatures in large networks of thermal capacitances and conductor. One such network represents a spent-fuel assembly in a core component pot surrounded by the EVTM cold wall. The program has been used to supplement thermal analyses performed with the TAP code.

Reference

"Manual for the Differential Equation Analyzer Program (DEAP)," B. L. McFarland, Rocketdyne Report LAP 69-552 (RC), October 1, 1969. (Rockwell International non-proprietary internal documentation.)

A.20 DEBLIN2 (Westinghouse Proprietary)

DEBLIN2 is a Westinghouse proprietary computer program which generates an earthquake motion having a response spectrum which closely resembles that of a given design response spectrum. The program consists of several subroutines which compute the response spectrum of a specified vibratory motion. After the response spectrum has been computed, another portion of the program compares it with the design response spectrum and determines at what frequency values spectrum modifications are necessary. Subroutines using the techniques of spectrum suppression and amplification make the appropriate changes in the time history, and the program returns to the spectrum - computation portion with the new time history. This process is repeated until a time history with an acceptable response spectrum is determined. The required modifications to the time history may also be made manually at each desired frequency.

Availability

The DEBLIN2 code is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

Verification of the capabilities of DEBLIN2 to correctly interpolate acceleration time histories and numerically integrate accelerations to obtain velocities and displacements, has been performed by comparison with closed-form solutions. Other verification problems performed consist of verifying the capability of the code to determine initial conditions to eliminate drift and the adequacy of the user selection of frequencies for spectrum modifications. Documentation of this verification is contained in a Westinghouse proprietary document (WCAP 8867).

Application

The DEBLIN2 code is used to generate synthesized time histories which are consistent with artificially broadened and smoothed design response spectra.

References

1. Tsai, N.-C., "Spectrum Compatible Motions for Design Purposes", Journal of of the Engineering Mechanics Division, Proceedings of the American Society of Civil Engineers, Vol. 93, No. EM2, April, 1972, pp. 345-356.

2. Johnson, N.L. and Leone, F. C., Statistics and Experimental Design in Engineering and the Physical Sciences, Vol. 1, John Wiley and Sons, 1964, pp. 384-385.

A.21 DEMO

The DEMO program is a digital computer simulation program designed to generate thermal-hydraulic transients for the CRBRP nuclear steam supply system. While the plant model is a three-loop calculation, the thermal hydraulic calculations of two of the loops are identical with the third loop representing a single loop where a perturbation might occur. Optionally, the hydraulics for the identical (lumped) loops can be modelled separately for cases such as a check valve closure in which the third loop thermodynamic effects are not required.

The reactor model is based on nuclear points Kinetics equations with six delayed neutron groups.

Core (fuel assemblies) thermodynamic modelling uses five axial nodes plus nodes for the lower and upper axial blankets for a total of seven (7). Each axial node has a sodium, one clad and two fuel nodes with added nodes for the duct and wire wrap. The radial blanket modelling includes seven axial nodes each of which includes a sodium, one clad and four fuel nodes, with additional nodes for the duct wire wrap. Doppler and coolant temperature feedbacks are simulated. Optionally, the radial expansion and assembly bowing feedback can be included. Dynamic models are included for the average, peak and hot channels. Only single-phase sodium flow is modelled. Tabular decay heat data is provided.

Loop piping simulation uses a variable-flow digital-storage transport delay model. Pipe metal heat storage is not represented (typically conservative for the cases simulated).

Loop hydraulics simulation includes friction and form losses along with momentum drops and elevation heads. Forced and natural circulation are simulated. The model for the coolant pumps include inertia and friction terms with fitted pump characteristics (head, torque and RNPSH). Pump cavitation is modelled for the pipe rupture simulation.

Each loop model includes a steam generator with superheater, evaporator (two modules lumped as one), steam drum and recirculation pump. The superheater and evaporator modules are each represented by seven heated nodes and water/steam. Heat transfer representationis available for liquid convection, nucleate boiling, transition boiling, film boiling and superheat.

In addition, a specialized steam generator program for blowdown and steam/feed pipe break analyses is included as an option. It is used to represent the DEMO S-loop steam generator when these analyses are required.

Simulation is provided for steam connections to a main steam header model, with the turbine load represented via pressure-dependent steam flow equations. Other turbine dynamics are currently not represented. Feed temperature is determined as a function of steam load.

Availability

DEMO is available as a CDC-7600 program at ARD. DEMO Rev. 3 and DEMO Rev. 4 are available through the Argonne Code Center.

Verification

DEMO Code analyses are performed in a number of categories for Design Thermal Transients and for Safety Analysis for CRBRP. Consequently, verification work has been and will be performed in corresponding categories.

1. DEMO Code steady-state operating conditions are set up to correspond to independently-calculated heat balances generated by hand calculations and/or steady-states analysis codes.

2. DEMO Code hydraulics have been compared with the Westinghouse proprietary IANUS Code as an independent analysis for the primary pipe rupture event.

3. The DEMO Code reactor response has been checked against the FORE-II reactor code response for the primary pipe rupture event.

4. The DEMO Code reactor response has been compared against the FORE-II reactor response for a plant trip transient.

5. ANL compared results of a DEMO core thermal hydraulic model to a more detailed calculation. The agreement was excellent leading to the conclusion that the DEMO Reactor Model is sufficient to describe the transient of the hot spot temperature and its location for a loss of flow transient. (ANL-CT-75-23).

The FFTF project has compared results of the IANUS Code with the SEFOR natural circulation tests. This IANUS verification supports the above DEMO - IANUS comparison.

Application

DEMO has been used for CRBRP transient design and safety analysis where a single-phase sodium model can be applied. It is also used to define input boundary conditions for more detailed subsystem analyses. Coverage includes:

(1) Normal plant operation
(2) Reactivity incidents
(3) Pump failure or seizure
(4) Natural Circulation
(5) Primary sodium pipe rupture
(6) Steam pipe rupture
(7) Other abnormal plant events, which may be represented by ad-hoc coding changes (e.g., loss of intermediate sodium).

Reference

"Clinch River Breeder Reactor Plant Nuclear Island. Simulation MOdel DEMO", WARD-D0005, Rev. 4, Westinghouse Advanced Reactors Division, Madison, PA., January 1976.

A.22 DOTIIIW or DOTIII

Discrete Ordinates Transport DOTIIIW or DOTIII solves the two-dimensional, Boltzmann transport equation with general anisotropic scattering for X,Y; R,Z; and R,θ geometries by using diamond difference or weighted difference solution techniques. DOTIIIW also solves the diffusion equation with procedures identical to the 2DB code.

DOTIIIW solves forward or adjoint homogeneous, or inhomogeneous problems. The program has a capability of using a fixed volume distributed source; a specified angular dependent boundary source at the left, right, top or bottom boundaries; or an interior radial or axial boundary source. Vacuum, reflective, periodic, white, or albedo boundary conditions may be specified. Time absorption calculations, concentration searches, or zone thickness searches are also solved; fissions may be included for a sub-critical system. Cross sections may be input from a library tape and/or from cards. Asymmetric or symmetric quadrature calculation may be performed. The code includes a choice of Gaussian Iteration, Successive Over-relaxation, Space Point Scaling, or Chebychev Acceleration to accelerate a flux solution on inner iterations.

Availability

The DOTIIIW code is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center. The current version was released from WANL in August, 1970, and updated by the version released from ORNL in September, 1973. DOTIIIW has been updated at ARD to satisfy CRBRP shield design analysis requirements.

Verification

Results of the DOTIIIW or DOTIII solution of the discrete ordinates transport and diffusion equations have been compared to exact analytic solutions, experimental data, and similar code results. Calculations performed with DOTIIIW or DOTIII have been compared with other two-dimensional code solutions and with experimental results. This has been done by numerous organizations for their particular version of the DOT code. In addition, a series of test cases is maintained on tape, and is run periodically to verify consistency of results. Hand calculations are done when applicable to verify minor updates required for CRBRP shield design analysis needs. Continuing documentation of the verification of the DOTIIIW or DOTIII solution of the discrete ordinates transport and diffusion equations is an extensive process due to the complexity of the DOT type of solution and the multiple options of the code.

Application

DOTIIIW or DOTIII is the principal shielding design analysis method employed on the CRBRP project. The various applications include: 1) prediction of in-vessel neutron and gamma flux environments for use in nuclear heating, material radiation damage studies, radiation source, decay power, and shielding performance studies, 2) prediction of neutron and gamma streaming in the complex geometries of the reactor cavity, reactor vessel support area, head access area, and primary heat transport pipe chaseway, 3) prediction of bulk shielding performance of the enclosure system shields (e.g., closure head assembly), 4) prediction of equipment neutron and gamma environments and shielding, (e.g., the flux monitor system), and 5) detailed analysis of the CRBRP shielding experimenta of radiation heating, material shielding worth, and basic neutron and gamma cross section measurements.

References

1. R. G. Soltesz, R. K. Disney, J. Jedruch, and S. L. Zeigler, "Nuclear Rocket Shielding Methods, Modification, Updating and Input Data Preparation, Volume 5: Two-Dimensional, Discrete Ordinates Transport Technique," WANL-PR-(LL)-034, August 1970.

2. W. A. Rhoades and F. R. Mynatt, "The DOTIII Two-Dimensional Discrete Ordinates Transport Code," ORNL-TM-4280, September 1973.

A.23 DRIPS (Rockwell International Proprietary)

DRIPS is a finite element computer program developed at Atomics International. It is designed to perform load analyses of three-dimensional linear elastic frame type structures including piping systems. DRIPS is not descendant from any other program.

Availability

Two versions of this program are currently available; DRIPS (normal sized problems) and DRIPS BIG (extremely large problems). A nonlinear version of this code is presently being utilized by Argonne National Laboratories.

Verification

The verification of DRIPS is presented in Reference 1. The PIPOYN dynamic problem and the AOLPIPE static pipe network problem were selected by General Electric for solutionby DRIPS to verify the suitability of DRIPS for use in CRBRP structural analysis per SRP 3.9.1, Section II.2.b.

Application

The DRIPS Code is used for structural analysis of three-dimensional frame structures of the Fuel Handling System components, Nak to air heat exchangers, the steam generator module, and the large component cleaning vessel. The code uses consistent mass matrices to predict natural frequencies, mode shapes, seismic loads and deflections, and reactions due to concentrated forces, inertial loads (gravity), temperature, and imposed displacements.

Reference

"Verification of DRIPS," E. C.Signorelli, Atomics International Report Document Number N03611639, February 2, 1976.

A.24 DUNHAM's

DUNHAM's is a finite ring element stress analysis program. It will determine the stresses and displacements of axisymmetric structures of arbitrary geometry subjected to either axisymmetric loads or non-axisymmetric loads represented by a Fourier series. This program is similar to the GASP program. The major differences are that DUNHAM's can handle non-axisymmetric loads (which requires that each node have three degrees of freedom) and the material properties for DUNHAM's must be constant. As in GASP, the loadings may be thermal, mechanical, and accelerational.

All applied loads, displacements and temperature distribution can be made non-axisymmetric by specifying them as terms of a Fourier series. The number of both symmetric and anti-aymmetric harmonics is input along with the coefficient of each harmonic. The coefficients for loads, displacements, and temperatures may all be different.

Chicago Bridge and Iron Company's version of DUNHAM's has been expanded from the original version to include two special features beyond the basic solution described above. These features include the calculation of stresses on the surface of the model and the calculation of equivalent shell type (membrane and membrane plus linear bending) stresses.

Availability

This program is available on the IBM 370, Model 165 computer of the Chicago Bridge and Iron Company.

Verification

Verification of this program is achieved through the use of the following problems.

1. Cantilevered Cylinder with Partially Distributed Load

   The purpose of this problem is to demonstrate DUNHAM's ability to determine stresses and deflections of a body due to a partially distributed (non-axisymmetric) load. The problem considered is a 42 inch O.D. by 38 inch I.D. cylinder, 60 inches long. The loading considered is an external pressure of 1000 psi, between X=48 inches and Z-54 inches and from 90° to 270°. This partially distributed load is shown in figure A.24-1.

   The DUNHAM's results are compared to results obtained from KALNINS program. The deflection comparisons are shown in figures A.24-2, A.24-3, A.24-4 and the axial stresses at $\theta=90^{o}$ are shown in figure A.24-5.

2. Cantilevered Cylinder with Thermal Gradient

The purpose of this problem is to demonstrate DUNHAM's ability to determine stresses and deflections due to thermal loads. The problem consists of a 42" O.D. by 38" I.D. cylinder 60" long. The inside surface of the cylinder is at $0^{o}F$ while the outside surface is at $400^{o}F$. The temperature varies linearly through the thickness. The cylinder is fixed at one end to prevent deflection and is free at the other end.

The results are compared to results obtained from KALNINS program. Deflection comparisons are tabulated in table A.24-1 and the stress comparisons are shown in table A.24-2. The output data locations are not identical for both programs for comparison against DUNHAM's results; it is necessary to average the KALNINS results across the layer thickness. Although this was done, the stresses are still in good agreement.

3. Cantilevered Cylinder with Axial Acceleration

The purpose of this problem is to illustrate DUNHAM's ability to determine stresses and deflections due to applied body forces. The problem consists of a 40 inch O.D. by 38 inch I.D. cylinder which is 60" long. The cylinder is loaded axially with an acceleration of 100g. The cylinder is fixed at one end to prevent deflection and is free at the other.

DUNHAM's results are compared to results obtained by KALNINS program. The deflection comparisons are shown in Table A.24-3 and the stresses are tabulated in table A.24-4.

4. Partial Spheres with External Pressure

The purpose of this problem is to illustrate DUNHAM's ability to determine stresses and deflections of a body due to surface loadings. The problem considered is an 84 inch O.D. by 76" I.D. sphere subjected to an external pressure of 10,000 psi.

The DUNHAM's results are compared to the elasticity solution obtained from theoretical formula from Roark, Young, "Formulas for Stress and Strain," Fifth Edition, McGraw Hill Book Co. page 506. The deflections are listed in table A.24-5 and the stresses are tabulated in table A.24-6.

5. Cylinder with Internal Pressure

The purpose of this problem is to illustrate DUNHAM's ability to determine stresses and deflections due to surface loadings on a thick-walled body of revolution. The problem considered is a 24 inch O.D. by 6 inch I.D. cylinder 3 inches long subjected to an internal pressure of 10,000 psi.

The results are compared to solutions obtained from theoretical equations for thick wall cylinders taken from Timoshenko, "Strength of Materials Part II," Third Edition, Van Nostrand Reinhold Co. page 210. The displacement comparisons are shown in table A.24-7. DUNHAM's results are in fair agreement with the theoretical results. Part of the error is due to the fact that the end nodes of the model were fixed in the Z direction, while the theoretical solution allows the ends to move in the axial direction. The stresses are shown in table A.24-8.

Application

DUNHAM's will be used in the finite element analysis of the various penetration in the CRBRP Head Assembly.

TABLE A.24-1

COMPARISON OF DEFLECTION RESULTS

Location		R Deflection			Z Deflection		
R	Z	DUNHAMS	KALNINS	% Diff	DUNHAMS	KALNINS	% Diff
20	0	0	0	0	0	0	0
20	10	.029253	.02823	3.6	.017419	.01734	0.5
20	20	.031599	.03071	2.9	.032072	.03224	0.5
20	30	.030905	.03001	3.0	.046812	.04721	0.8
20	40	.030870	.03007	2.7	.061569	.06219	1.0
20	50	.027116	.02586	4.9	.076561	.07742	1.1
20	60	.051140	.05360	4.6	.091259	.09221	1.0

TABLE A.24-2

COMPARISON OF STRESS RESULTS

Location		Z-Stress			T-Stress		
R	Z	DUNHAMS	KALNINS	% Diff	DUNHAMS	KAKNINS	% Diff
20.75	20.50	47020	47882	1.8	48570	49057	1.0
20.25	20.50	15010	15961	6.0	17310	16983	1.9
19.75	20.50	-17020	-15964	6.6	-14710	-15095	2.6
19.25	20.50	-49030	-47885	2.4	-47560	-47173	.8
20.75	30.5	47000	48018	2.1	47690	47995	.6
20.25	30.5	15000	15996	6.2	16410	16076	2.1
19.75	30.5	-17000	-15999	6.3	-15640	-16024	2.4
19.25	30.5	-49000	-48000	2.1	-48520	-48122	.8

TABLE A.24-3

COMPARISON OF DEFLECTION RESULTS

Location		R Deflection			Z Deflection		
R	Z	DUNHAMS	KALNINS	% Diff	DUNHAMS	KALNINS	% Diff
20	0	.0003333	.0003332	0	0	0	-
20	10	.00029226	.0002920	0.1	-.00053082	-.0005306	0.1
20	20	.00023171	.0002316	0.1	-.00096528	-.0009650	0
20	30	.00017374	.0001737	0	-.0013031	-.001303	0
20	40	.00011586	.0001158	0.1	-.0015159	-.001544	1.8
20	50	.000057918	.00005790	0	-.0016891	-.001689	0
20	59	.000005934	.000005789	2.5	-.0017369	-.001737	0

TABLE A.24-4

COMPARISON OF STRESS RESULTS

Location		Z-Stress			T-Stress		
R	Z	DUNHAMS	KALNINS	% Diff	DUNHAMS	KALNINS	% Diff
20.75	10.50	-1431	-1432	0.1	3.938	3.803	3.5
20.25	10.50	-1432	-1432.5	0	3.757	3.589	4.7
19.75	10.50	-1433	-1433.5	0	3.551	3.374	5.2
19.25	10.50	-1434	-1434	0	3.354	3.16	6.1
20.75	40.5	-564.2	-564.5	0.1	NEGLIGIBLE		
20.25	40.5	-564.2	-564.5	0.1			
19.75	40.5	-564.2	-564.5	0.1			
19.25	40.5	-564.3	-564.5	0.1			

TABLE A.24-5

RADIAL DEFLECTIONS

R	ELASTICITY SOLUTION	DUNHAMS NODE	DUNHAMS SOLUTION	% DIFF
38	-.04757640	1	-.047916	0.7
39	-.04850196	2	-.048696	0.4
40	-.04942752	3	-.049569	0.3
41	-.05035308	4	-.050546	0.4
42	-.05127864	5	-.051635	0.7

TABLE A.24-6

STRESSES - ELASTICITY SOLUTION

R	Z	σ_R	σ_Z	σ_T	σ_{RZ}
1.81	41.42	-53270	-9040	-53355	1944
1.77	40.42	-54387	-6800	-54479	2082
1.72	39.42	-55620	-4327	-55719	2244
1.68	38.43	-56985	-1590	-57092	2423

STRESSES - DUNHAMS Results

R	Z	σ_R	% DIFF	σ_Z	% DIFF	σ_T	% DIFF	σ_{RZ}	% DIFF
1.81	41.42	-53680	.8	-9088	.5	-53760	.8	2323	20
1.77	40.42	-54570	.3	-6824	.4	-54650	.3	2530	21.5
1.72	39.42	-55610	0	-4319	.2	-55700	0	2708	20.7
1.68	38.43	-56740	.4	-1580	.6	-56830	.5	2850	17.6

TABLE A.24-7

RADIAL DEFLECTIONS

R	Z	ELASTICITY SOLUTION	DUNHAMS	% DIFF
3	1.5	.0013866	.0014113	1.8
4	1.5	.0011022	.0010796	2.1
5	1.5	.00090977	.0008847	2.8
6	1.5	.0007866	.00075849	3.6
7	1.5	.00070317	.00067157	4.5
8	1.5	.0006444	.00060923	5.8
9	1.5	.00060222	.00056328	6.5
10	1.5	.00057155	.00052882	7.5
11	1.5	.00054929	.0005027	8.5
12	1.5	.0005333	.00048286	9.5

TABLE A.24-8

COMPARISON OF STRESS RESULTS

R	R-STRESS			T-STRESS		
	ELASTICITY SOLUTION	DUNHAMS	% DIFF	ELASTICITY SOLUTION	DUNHAMS	% DIFF
3.5	7170	7233	0.9	8503	8623	1.4
4.5	4074	4089	0.4	5407	5438	0.6
5.5	2507	2509	0.1	3840	3846	0.2
6.5	1606	1605	0.1	2939	2936	0.1
7.5	1040	1039	0.1	2373	2368	0.2
8.5	662	661	0.2	1996	1989	0.4
9.5	397	396	0.3	1730	1724	0.3
10.5	204	204	0	1537	1531	0.4
11.5	59	60	1.7	1393	1386	0.5

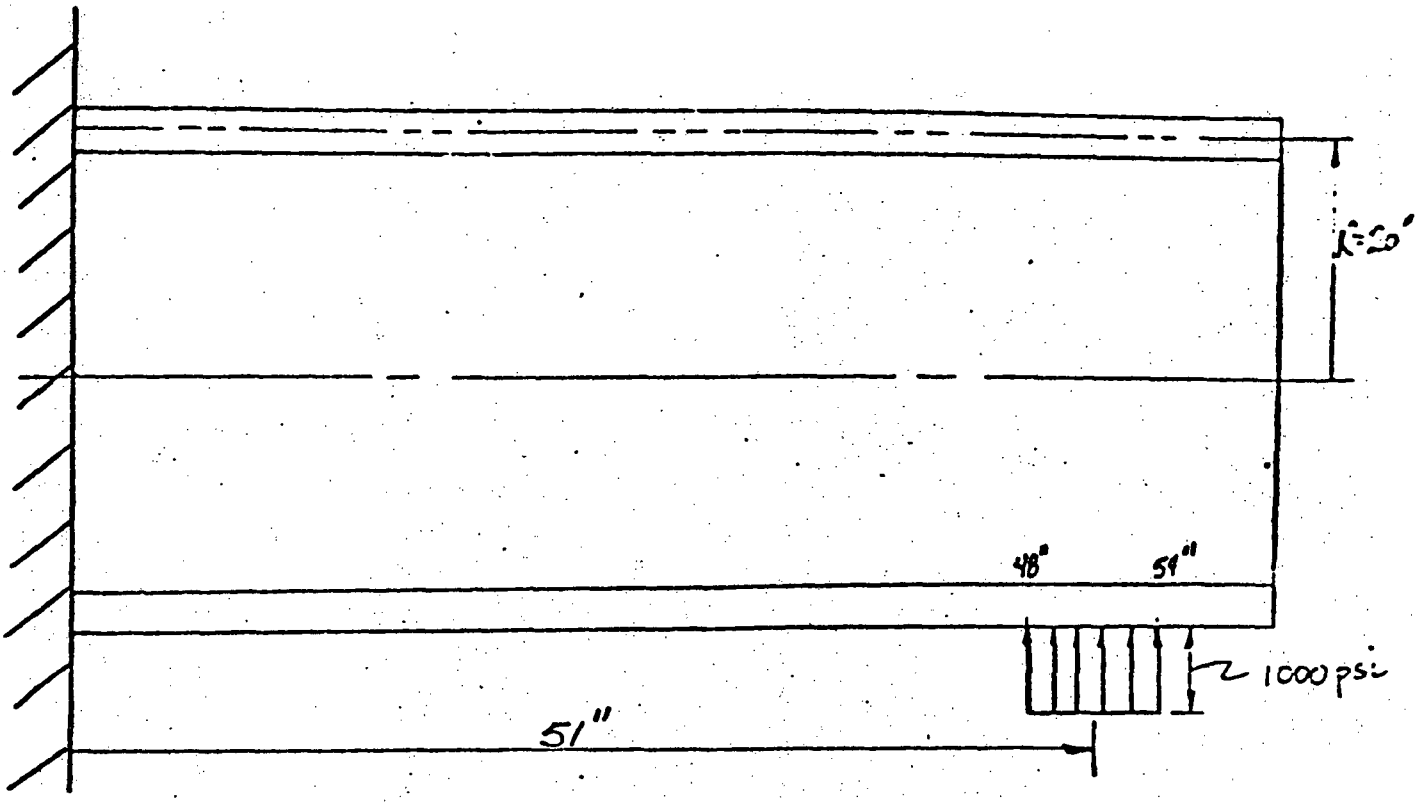
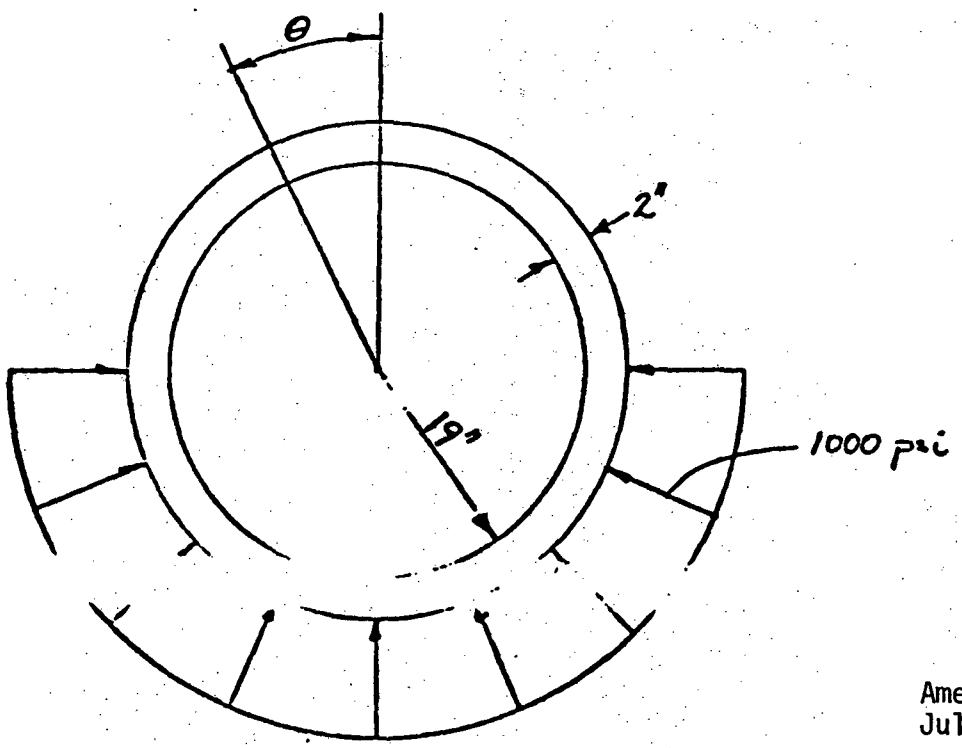


FIGURE A.24-1

CANTILEVERED CYLINDER UNDER PARTIALLY DISTRIBUTED LOAD



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July 1978

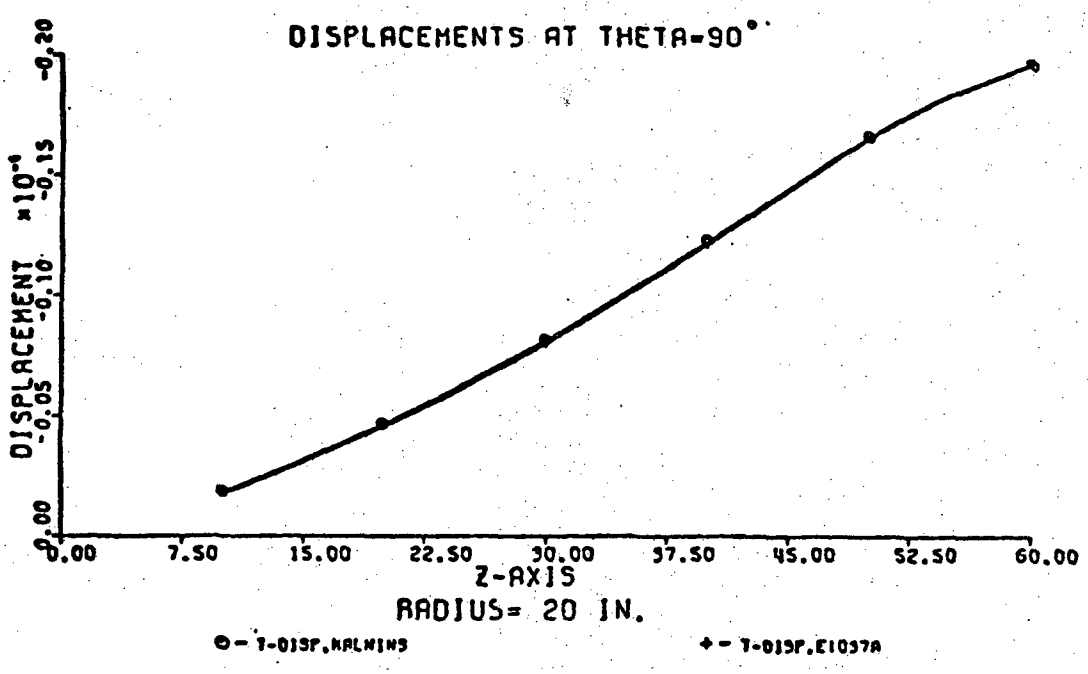
DISPLACEMENTS AT THETA=90°
DISPLACEMENT ×10⁻⁴
0.00
-0.05
-0.10
-0.15
-0.20
0.00
7.50
15.00
22.50
30.00
37.50
45.00
52.50
60.00
Z-AXIS
RADIUS= 20 IN.
⊙ - T-DISP,NALWINS
+ - T-DISP,E1037A

FIGURE A.24-2

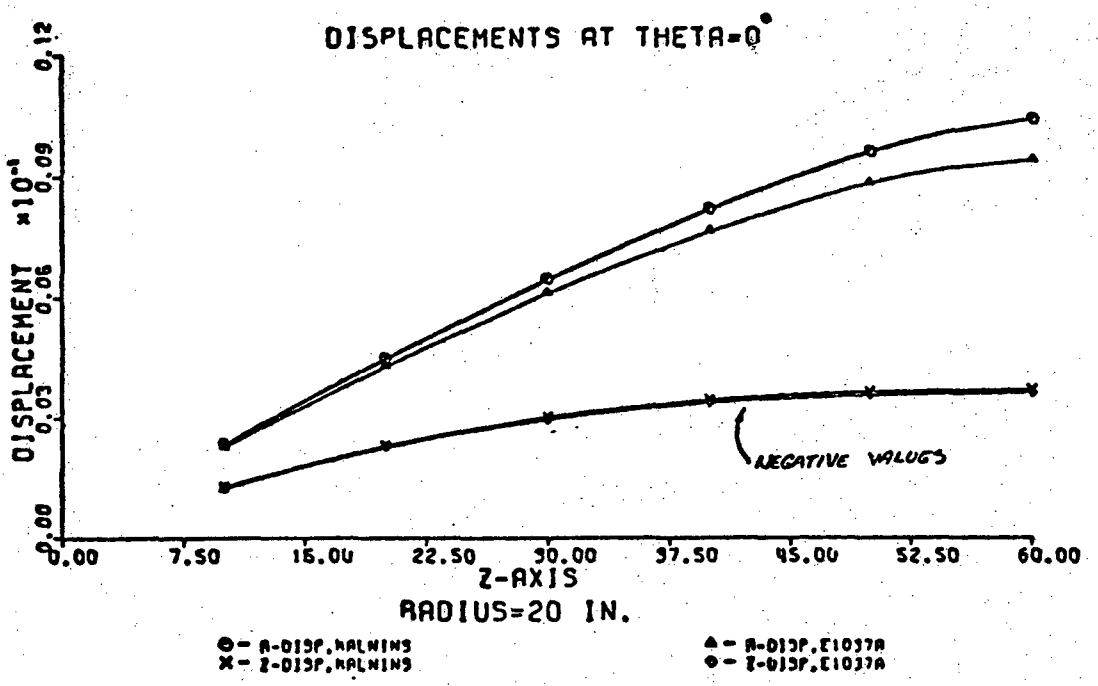
DISPLACEMENTS AT THETA=0°
DISPLACEMENT ×10⁻¹
0.00
0.03
0.06
0.09
0.12
0.00
7.50
15.00
22.50
30.00
37.50
45.00
52.50
60.00
Z-AXIS
RADIUS=20 IN.
NEGATIVE VALUES
⊖ - R-DISP,NALWINS
× - Z-DISP,NALWINS
▲ - R-DISP,E1037A
⊕ - Z-DISP,E1037A

FIGURE A.24-3

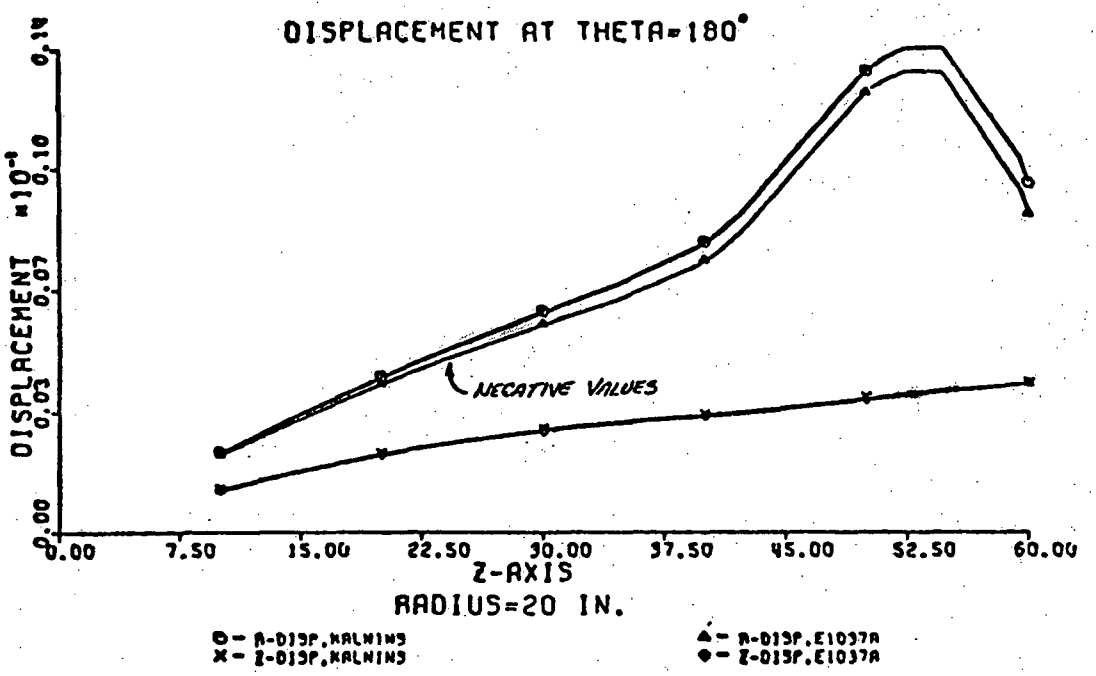
DISPLACEMENT AT THETA=180°
DISPLACEMENT ×10⁻³
0.00
0.03
0.07
0.10
0.14
NEGATIVE VALUES
0.00
7.50
15.00
22.50
30.00
37.50
45.00
52.50
60.00
Z-AXIS
RADIUS=20 IN.
O - R-DISP,KALNING
X - Z-DISP,KALNING
▲ - R-DISP,E1037A
● - Z-DISP,E1037A

FIGURE A.24-4

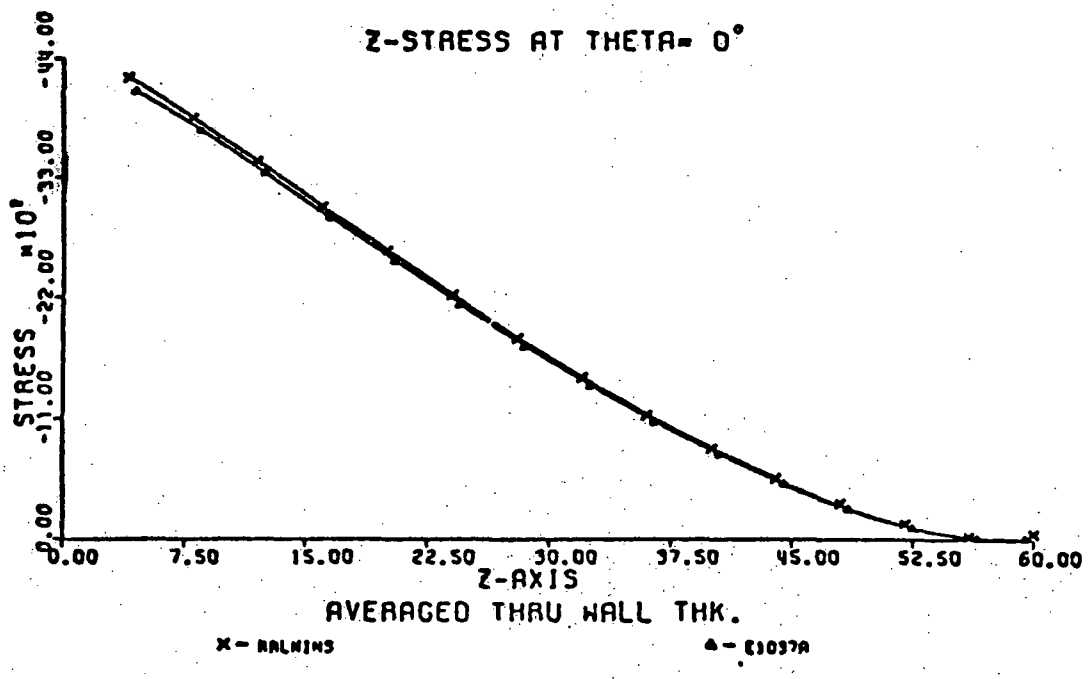
Z-STRESS AT THETA= 0°
STRESS ×10⁸
0.00
-11.00
-22.00
-33.00
-44.00
0.00
7.50
15.00
22.50
30.00
37.50
45.00
52.50
60.00
Z-AXIS
AVERAGED THRU WALL THK.
X - RALKINS
Δ - E1037A

FIGURE A.24-5

A.25 DYNALSS

DYNALSS code was written to compute the hydrodynamic scram response of the Secondary Control Rod System (SCRS). The pressures in the high and low pressure plenums and above the assembly are specified as a function of time and represent the boundary conditions on the control assembly. The code computes the steady-state flow and pressure distributions prior to scram. During the scram, the code calculates flows, pressures, velocity, and displacement as a function of time.

Availability

DYNALSS (April 1977 version) is available on the Honeywell 6000 computer system of General Electric in San Jose.

Verification

Validation of the DYNALSS code, per Standard Review Plan SRP Section 3.9.1.II.2.C, was discussed in a paper given at the 1974 Winter ANS Meeting (Reference). As of November 1982, DYNALSS was verified by SCRS Prototype-2 data. Verification by Prototypes-3 and -4 is planned prior to 1984.

Application

The DYNALSS code will be used to compute the behavior of the SCRS during various transients.

Reference

A. J. Lipps, "Scram Dynamics of a Control Rod with Hydraulic Scram Assist," American Nuclear Society Transactions, 1974 Winter Meeting, Washington, D.C., Volume 19, pg. 275-276.

A.26 DYNAPLAS (SAMMSOR IV and DYNAPLAS II)

DYNAPLAS is a finite element program for the dynamic, large deformation, elastic-plastic analysis of stiffened shells of revolution. It contains two programs: SAMMSOR IV is to determine stiffness and mass matrixes for curved shell element and or curved beam element; DYNAPLAS II is utilized for the analysis with an option of either the Houbolt or central difference numerical integration scheme being used to solve the equations of motion. DYNAPLAS II accepts as input a tape containing the structural stiffness and mass properties generated by SAMMSOR IV.

Availability

DYNAPLAS has been available at the CDC 7600 computer of Lawrence Berkeley Laboratory since August, 1975. The version was released by Texas A & M University in October, 1973.

Verification

DYNAPLAS has been extensively verified. The results predicted by DYNAPLAS are in close agreement with those from experimental programs (Refs. 1 and 2). See Figure A.26-1.

Application

DYNAPLAS will be applied to axisymmetric shell structures with or without ring stiffeners for elastic and inelastic dynamic analysis. It is currently used for studying rupture disc behavior.

References

(1) Haisler, W. E., and Stricklin, J. A., "SAMMSOR IV-A Finite Element Program to Determine Stiffness and Mass Matrices of Ring-Stiffened Shells of Revolution," TEES-2926-73-1, and SLA-73-1105, Texas A & M University, Texas, October, 1973.

(2) Haisler, W. E., and Vaughan, D. K. "DYNAPLAS II - A Finite Element Program for the Dynamic, Large Deflection, Elastic-Plastic Analysis of Stiffened Shells of Revolution," TEES-2926-73-2, and SLA-73-1106, Texas A&M University, Texas, October, 1973.

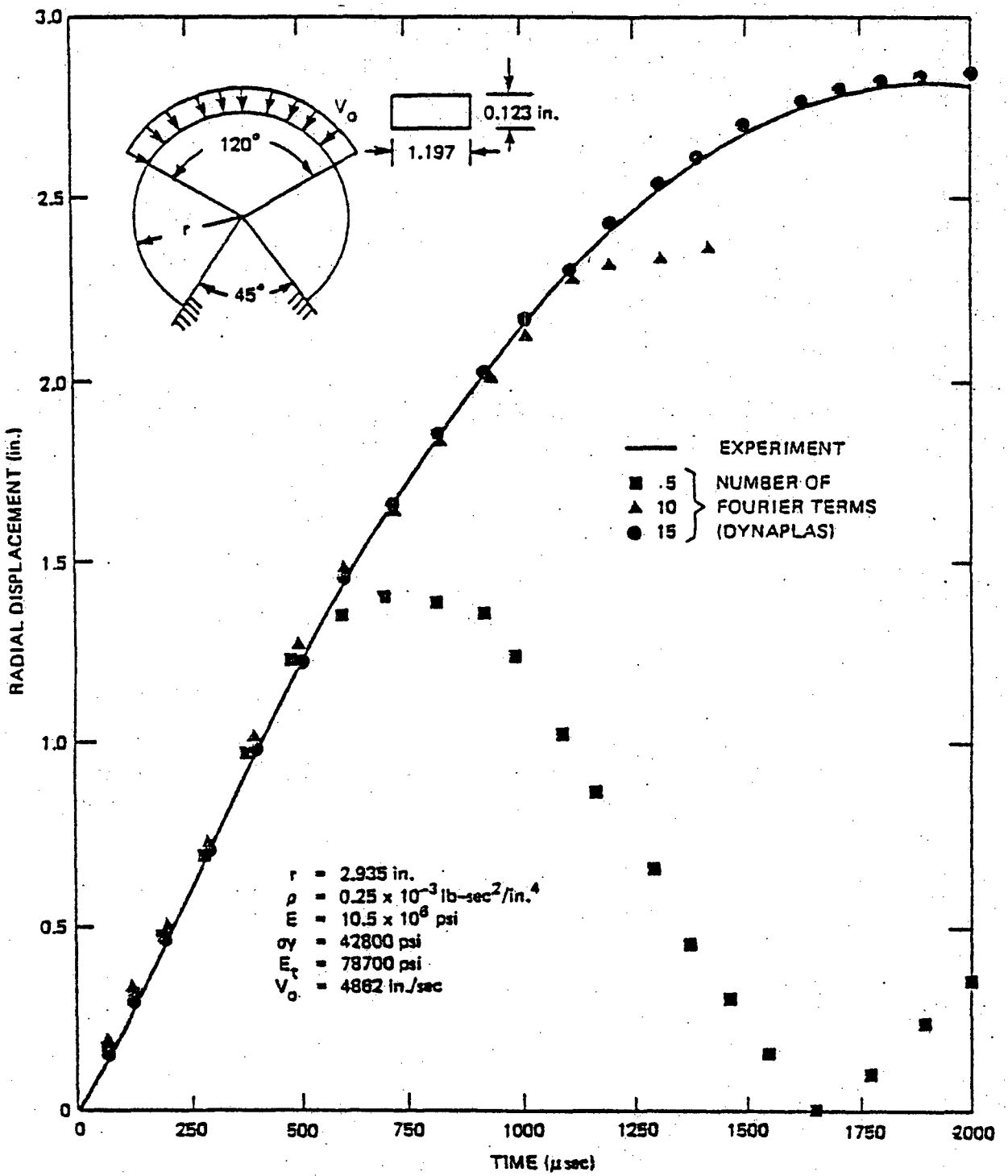

Figure A.26-1 *Comparison of DYNAPLAS with Experimental Results*

A.27 ELTEMP

ELTEMP is a post processing program which evaluates the results of the piping analysis for thermal expansion, dead weight, seismic, and thermal transient analysis according to the rules of Code Case 1331. This program has been modified to reflect the latest rules for High Temperature Design, Code Case 1592.

Availability

ELTEMP (Reference 1) is available on the CDC 7600 computer of Lawrence Berkeley Laboratory, Berkeley, California, and on the CDC 7600 computer of Westinghouse Power Systems at the Monroeville Nuclear Center.

Verification

The comparison with hand calculations and and other code results for Code Case 1331 is documented in References 1, 2 and 3. Extension of verification to Code Case 1592 is in progress.

Application

It will be used in the structural evaluation of ASME Class 1 elevated temperature piping.

References

(1) "ELTEMP - A Computer Program for Structural Evaluation of Elevated Temperature Piping", WARD-D-0069, February, 1975.

(2) "ELTEMP - A Computer Program for Structural Evaluation of Elevated Temperature Piping, Part 2, Appendices", WARD-D-0069, February 1975.

(3) "Manual for Using the ELTEMP Computer Program", Bechtel Corp., FFTF Stress Group, San Francisco 1973.

A.28 ETOX

ETOX (ENDF/B to 1DX) calculates group constants for nuclear reactor calculations from current available microscopic neutron data files such as the Evaluated Nuclear Data File.

Availability

ETOX is currently available at the Hanford Engineering Development Laboratory (HEDL) computer facility in Richland, Washington.

Verification

ETOX is a link for calculating neutron cross section data for both the plate geometry critical experiments and the pin geometry CRBRP core. These cross sections are then employed in nuclear analysis computer codes using reference calculational techniques to analyze supporting critical experiments to establish bias factors and uncertainties for criticality, reactivity coefficients, control rod worths, reaction rates, etc. These bias factors and uncertainties are then applied to the analysis of the CRBRP core, again using reference calculational techniques and cross sections generated, in part, by ETOX. This analytical technique is fully described in subsection 4.3.2 of the PSAR.

Application

The output from the ETOX code includes "infinite dilute" group cross sections, inelastic transfer matrices and temperature, as well as resonance dependent self-shielding factors for arbitrary values of the total cross section per absorber atom. The output from ETOX, in the appropriate format, is then input to the XSRES-W1DX code.

Reference

R. E. Schenter, J. L. Baker, and R. B. Kidman, "ETOX: A Code to Calculate Group Constants for Nuclear Reactor Calculations", BNWL-1002, May 1969.

A.29 E0984A

E0984A can be used to calculate the stress intensity of the stress differences, on a component level, between two different stress conditions.

The calculations can be done at as many as 20 points in space (elements) in one run. For any given element, the program can handle up to 40 different points in time or stress conditions. It considers the stress differences between every possible combination of two times.

E0984A can handle non-axisymmetric stresses, generally caused by mechanical loads. As many as 30 mechanical loads can be considered.

For each time combination and each mechanical load combination, the stress intensity of the stress difference components is calculated.

From all the calculations described above, the maximum values are retained and printed.

Any stress difference involves six stress components, three direct stresses and three shear stresses. The stress intensity is calculated by finding the three stress invariants from the six components, solving a cubic equation with the invariants as coefficients to get the three principal stresses, and finally finding the maximum absolute value of the algebraic difference between any two principal stresses. This value is equal to twice the maximum shear stress and is by definition, the stress intensity.

Availability

This program is available on the IBM 370, model 165 Computer of the Chicago Bridge and Iron Company.

Verification

Verification of this program is achieved through the use of the following problem.

The purpose of this problem is to illustrate E0984A's ability to determine the maximum range of stress intensity at a point due to pressure, mechanical and thermal stresses. The sample problem was prepared from data generated during the analysis of the coolant outlet nozzles of the 157 inch Westinghouse P.W.R. Figures A.29-1 and A.29-2 show a section through the model and the approximate location of the element (#501) being considered. The five mechanical loads which are applied to the nozzle are also shown in Figures A.29-1 and A.29-2. The program results and the hand calculation results are in exact agreement.

Application

Program E0984A will be used in the design of the closure head assembly to analyze the range of stress intensity (3 S_m).

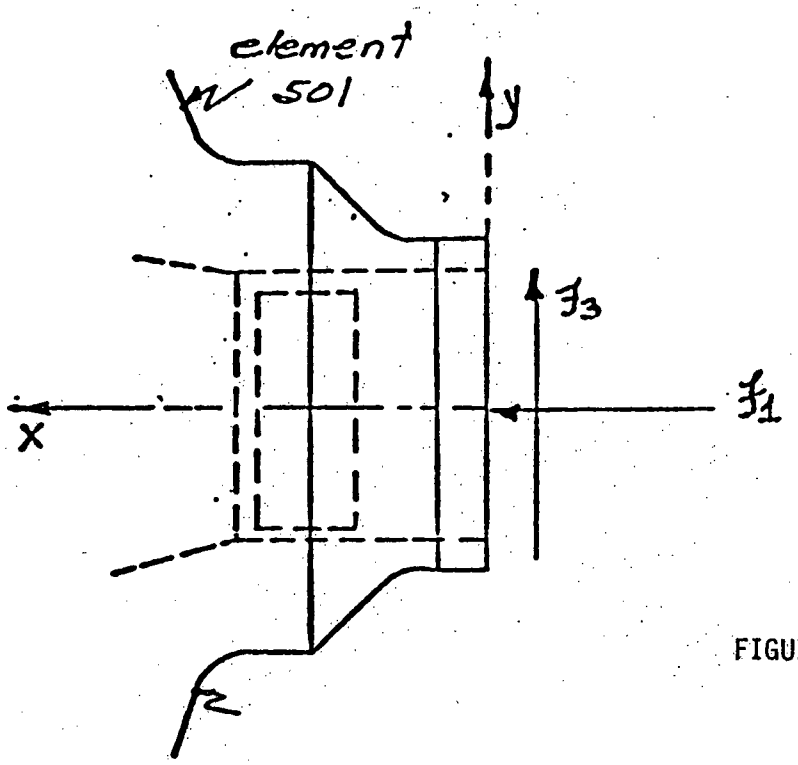

FIGURE A.29-1

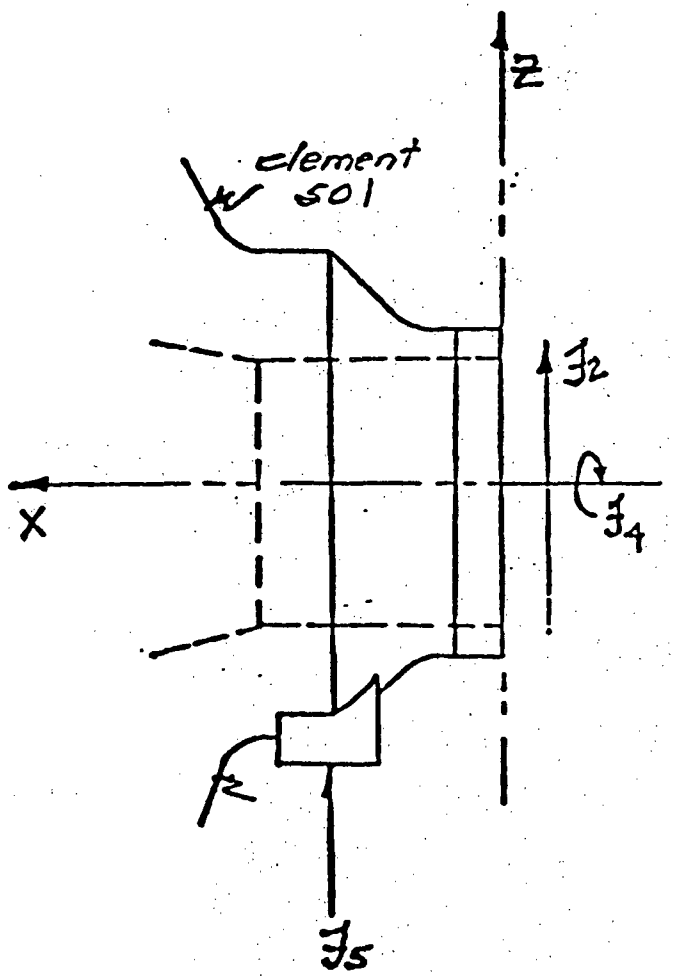

FIGURE A.29-2

A.30 E1682A

E1682A is a general purpose finite element program with beams, triangular flat plates, quadrilateral flat plates and rectangular plates. Both membrane and bending action is included in the plate elements.

Loads considered in the program are normal uniform loads on both the beam and plates and direct concentrated loads on nodes. Membrane and bending thermal stresses are also included for the triangular plates.

The beam element assumes a linear deflection along the beam axis and a cubic deflection pattern normal to the beam.

The triangular plate stiffness is a fully conforming element on the interior and on exterior edges. The convergence to a given problem is thus monotonic. It is a subdivided triangle similar to the Clough-Tocher element used in the NASTRAN program. The bending element is combined with a constant strain triangle.

The rectangular plate element uses a slightly higher order polynominal. In addition it is a nonconforming element which is integrated numerically. It will therefore be a softer element than the Clough element.

Availability

This program is available on the IBM 370, Model 165 Computer of the Chicago Bridge and Iron Company.

Verification

Verification of the program is achieved through the use of the following problems:

1. Spherical Shell

   The problem consists of a sphere under six equal loads each 90° apart from the other, see figure A.30-1, the loading is applied in the form of a normal pressure of 100 psi on the plate elements applied overa small circular area. The sphere has an inside radius of 100 inches and is two inches thick.

   The results of E1692A are compared to those of KALNINS. Plots of inside, outside, and meridional stresses are shown in figures A.30-2, A.30-3 and A.30-4 (longitudinal) and figures A.30-5, A.30-6 and A.30-7 (circumferential) for programs E1682A and KALNINS.

2. Thin Cylindrical Shell to Nozzle Junction

This problem makes a comparative study for the detailed stresses that occur at a thin cylindrical shell to nozzle junction using a finite element analysis and a shell analysis. An evaluation is then made of the program results with experimental test data. The three load cases considered are a) radial load on the nozzle, b) in plane moment on nozzle, and c) out-of-plane moment on the nozzle.

Comparison of results obtained from program E1682A (finite Element method), modified KALNINS and experimental tests are presented in figures A.30-8 through A.30-19 the firm lines denote results from program E1682A, the dashed lines denote results from KALNINS and experimental stress values are denoted by E. All of the plots are for stresses in the shell and should not be tied into stresses along the neck of the nozzle.

3. Circular Plate Modeled with Isoparametric Brick Elements

The problem consists of a circular plate simply supported along the periphery of the top edge which is 22.0 inches thick and has a radius of 27.845 inches. A uniform pressure of 1.000 psi is applied at the bottom surface as shown in figure A.30-20.

The program results for these elements are transformed to o, stresses and compared to the elasticity solution using thick plate theory reference Timoshenko, S. and Goodier, J. "Theory of Elasticity," second edition, McGraw Hill, 1951 page 351. The results are shown in table A.30-3.

4. Circular Disk with Constant Thermal Gradient

The problem consists of a 1 inch thick circular plate with a radius of 40 inches which has a constant thermal gradient. The temperature on the outside of the plate is +50°F and the temperature on the inside of the plate if -50°F.

Program E1682A's results are compared to the theoretical solution from Woinowsky-Krieger, S. and Timoshenko, S., "Theory of Plates and Shells," McGraw Hill, New York, 1959 page 59 as shown in table A.30-1.

5. Elliptical Cylinder with Triangular, Rectangular, and Quadrilateral Plate Elements

The problem consists of 1" thick elliptical cylinder which has a uniform pressure of 10 psi applied to the inside surface the geometry and boundary conditions for the ellipse are shown in figure A.30-21.

The stress results are plotted in the quadrilateral plus the triangular plate element model. These plots are shown in figures A.30-22, A.30-23 and A.30-24.

6. Right Circular Cylinder Modeled with Quadrilateral and Rectangular Elements

The problem consists of a 0.25 inch thick right circular cylinder which is subjected to a uniform internal pressure of 10 psi. The geometry is shown in figure A.30-25. Program E1682A results are compared to the classical solution from strength of materials

$$\sigma r = \frac{P_r}{t}$$

and is shown in table A.30-2.

7. Sphere with Two Circumferential Stiffnesses

The problem consists of a sphere with two circumferential stiffnesses on theoutside surface. A uniform pressure load of 10 psi is applied on the inside surface. The sphere has an inside radius of 200 inches and is 0.25 inches thick. The stiffnesses are located at Ø angles of 105° and 90° and extend on lines parallel to the spherical centerline.

E1682's results are compared to KALNINS as shown in table A.30-4 and figure A.30-26.

Application

Program E1682A will be used in the design of the closure head assembly for the stress report section for the small rotating plug and IVTM penetration, for thermal and stress analysis of skirts and for stress analysis of 10.75" column to shield plate attachments.

TABLE A.30-1

	E1682	Theoretical	% Diff
Wmax (case 1)	0.05205	0.05200	0.10%
V_{max} (case 2)	1,471 psi	1392.85psi	5.61%

TABLE A.30-2

	E1682A (psi)	Theoretical (psi)	% Diff
σ_θ	396.6	405	2.07
σ_ϕ	0	0	-

TABLE A.30-3

	ELASTICITY SOLUTION	PROGRAM RESULTS		ELASTICITY SOLUTION	PROGRAM RESULTS	
ELEMENT #	σ_r (psi)	σ_r (psi)	% DIFF.	σ_z (psi)	σ_z (psi)	% DIFF
223	-1,773.7	-1,557.9	12.15	-989.7	-984.76	0.5
84	-1,666.9	-1,406.93	15.60	-989.7	-975.16	1.37
203	[illegible]	[illegible]	3.77	[illegible]	[illegible]	3.31

The results show a maximum difference of 15.60% from the elasticity solution in the σ_r direction for element number 84. Part of this difference is caused by the fact that the radial stresses for the elasticity solution at the outside boundary of the plate are not zero, as they are in program E1682A, but are

$$(\sigma_r)_{r=a} = q\left(\frac{2+\nu}{8}\,\frac{z^3}{c^3} - \frac{3}{8}\,\frac{2+\nu}{5}\,\frac{z}{c}\right)$$

Because the plate is so thick in relation to its radius, this Saint-Venant effect does affect stresses in the interior of the plate.

TABLE A.30-4

COMPARISON OF STRESS RESULTS

Ø, DEG.	V(Ø) PSI MERIDIONAL		% DIFF.	V(Ø) PSI MERIDIONAL		% DIFF.
	E1682A	KALNINS		E1682A	KALNINS	
115	3995	3983.0	0.3	3996	4000.5	0.1
113	4005	4001.0	0.1	3999	4000.0	0.02
111	4060	4048.0	0.3	3998	4000.5	0.06
109	3870	4028.5	4.0	4005	4001.5	0.09
107	3500	3413.5	3.0	3940	3999	1.5
105	2445	2347.5	4.0	3850	3988	3.6
103	3300	3424.5	3.8	4080	4001	1.9
101	3900	4066.0	4.2	4030	3999.5	0.7
99	4020	4070.0	1.2	4040	4000.5	1.0
97	3490	3573.0	2.4	3900	3999.5	2.6
95	2885.5	2777.5	3.7	3879	3997	3.0
93	3480	3567.5	2.5	4050	4000	1.2
91	3900	4019.5	3.1	4050	4000	1.2
89	4030	4033.5	0.1	4000	4000	0.1
87	4000	3999.5	0.01	3998	4000	0.05
85	3994	3987.0	0.2	3997	4000	0.07

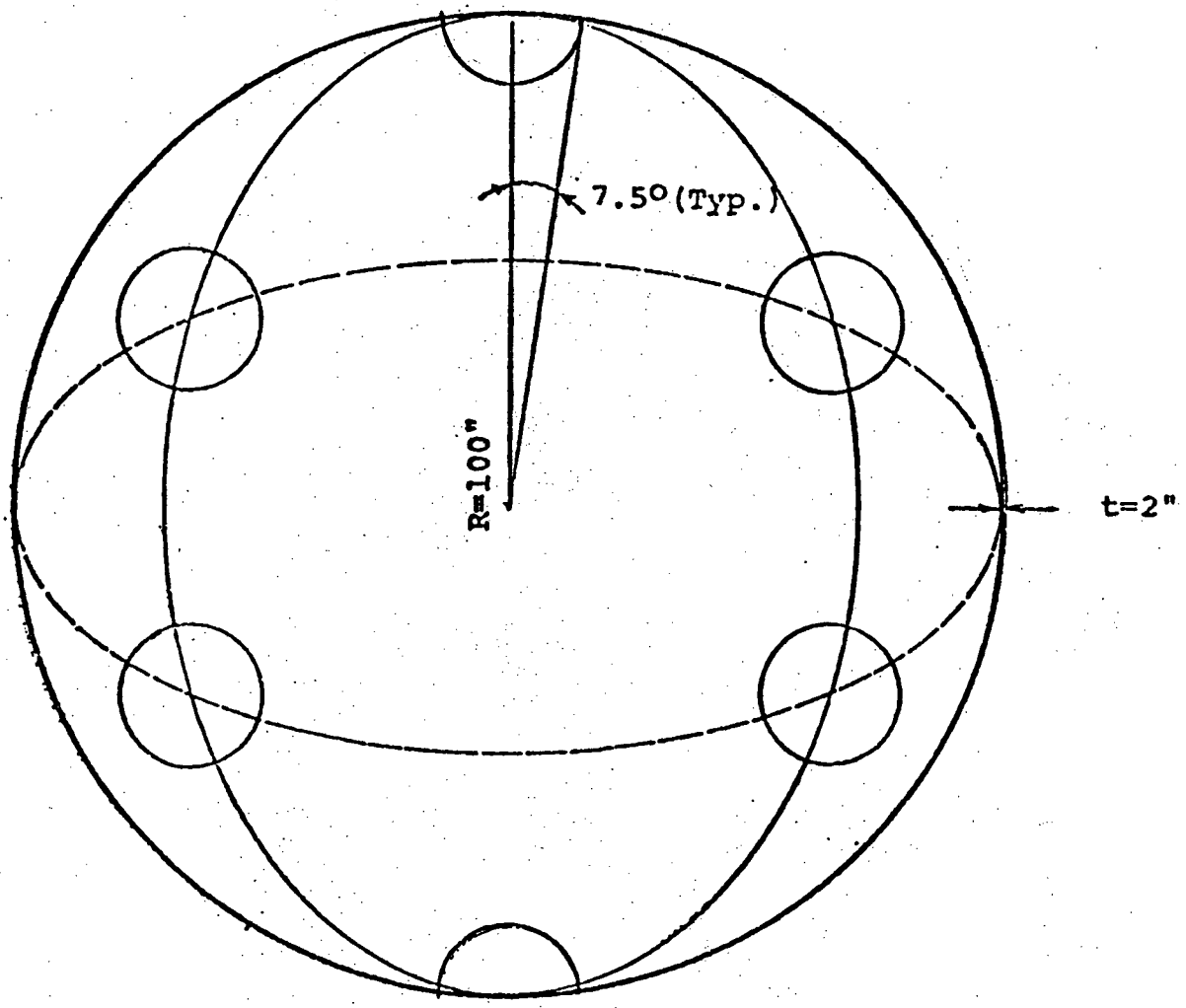

FIGURE A.30-1

GEOMETRY OF SPHERE

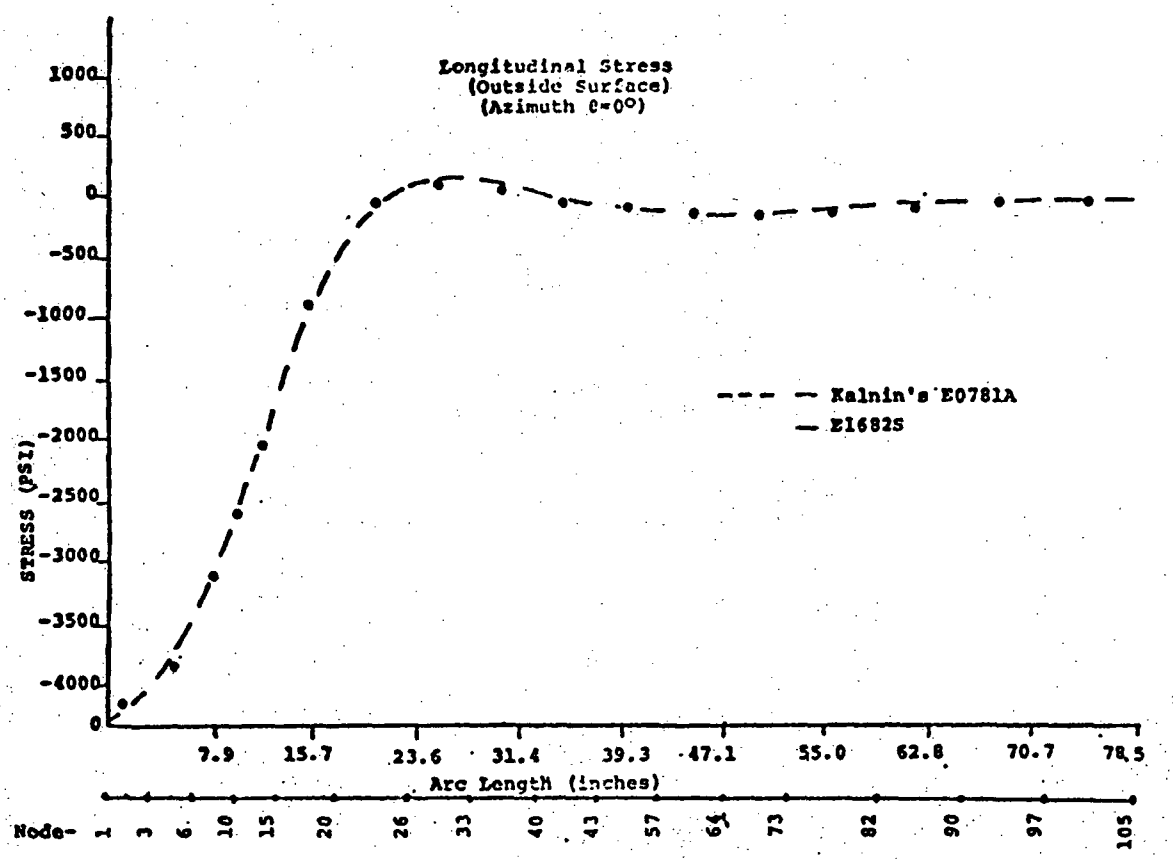
Longitudinal Stress
(Outside Surface)
(Azimuth θ=0°)
1000
500
0
-500
-1000
-1500
-2000
-2500
-3000
-3500
-4000
0
STRESS (PSI)
Kalnin's E0781A
E1682S
7.9
15.7
23.6
31.4
39.3
47.1
55.0
62.8
70.7
78.5
Arc Length (inches)
Node-
1
3
6
10
15
20
26
33
40
43
57
64
73
82
90
97
105

Figure A.30-2

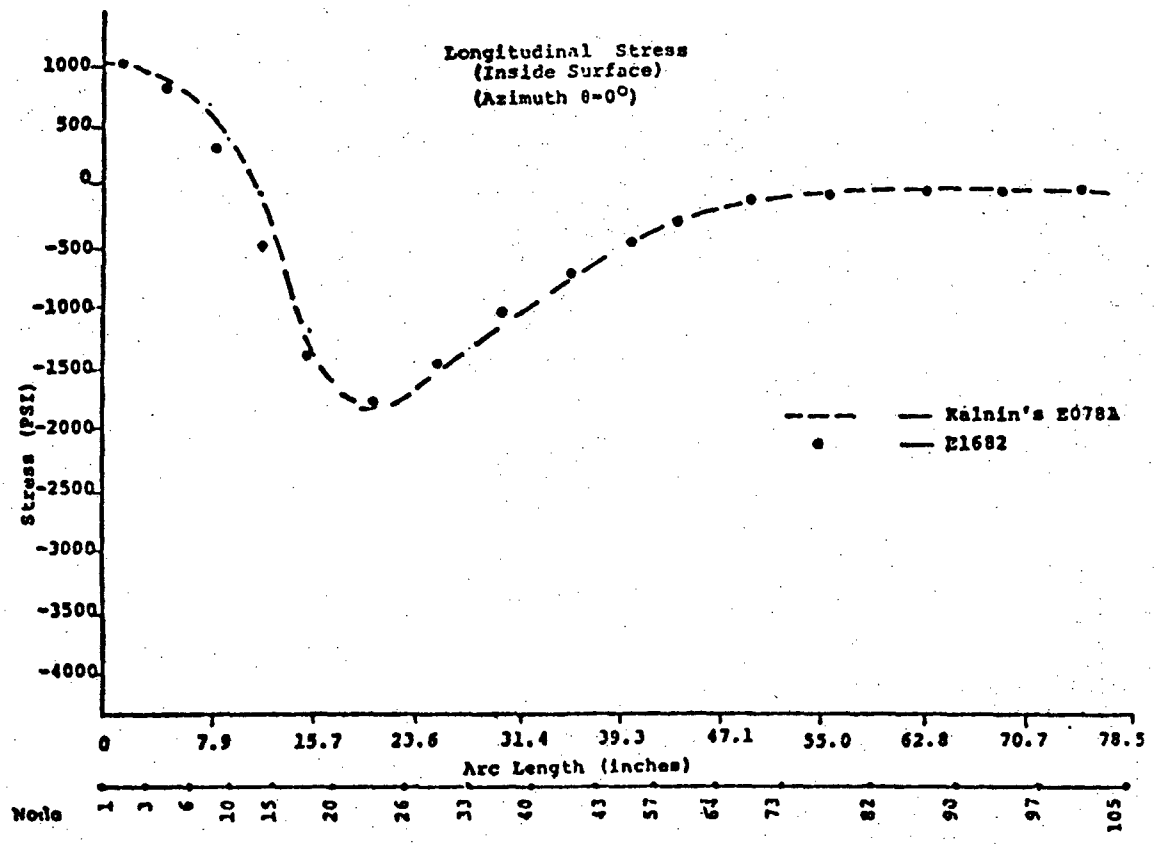
Longitudinal Stress
(Inside Surface)
(Azimuth θ=0°)
1000
500
0
-500
-1000
-1500
-2000
-2500
-3000
-3500
-4000
Stress (PSI)
Kalnin's E078A
E1682
0
7.9
15.7
23.6
31.4
39.3
47.1
55.0
62.8
70.7
78.5
Arc Length (inches)
Node
1
3
6
10
15
20
26
33
40
43
57
64
73
82
90
97
105

Figure A.30-3

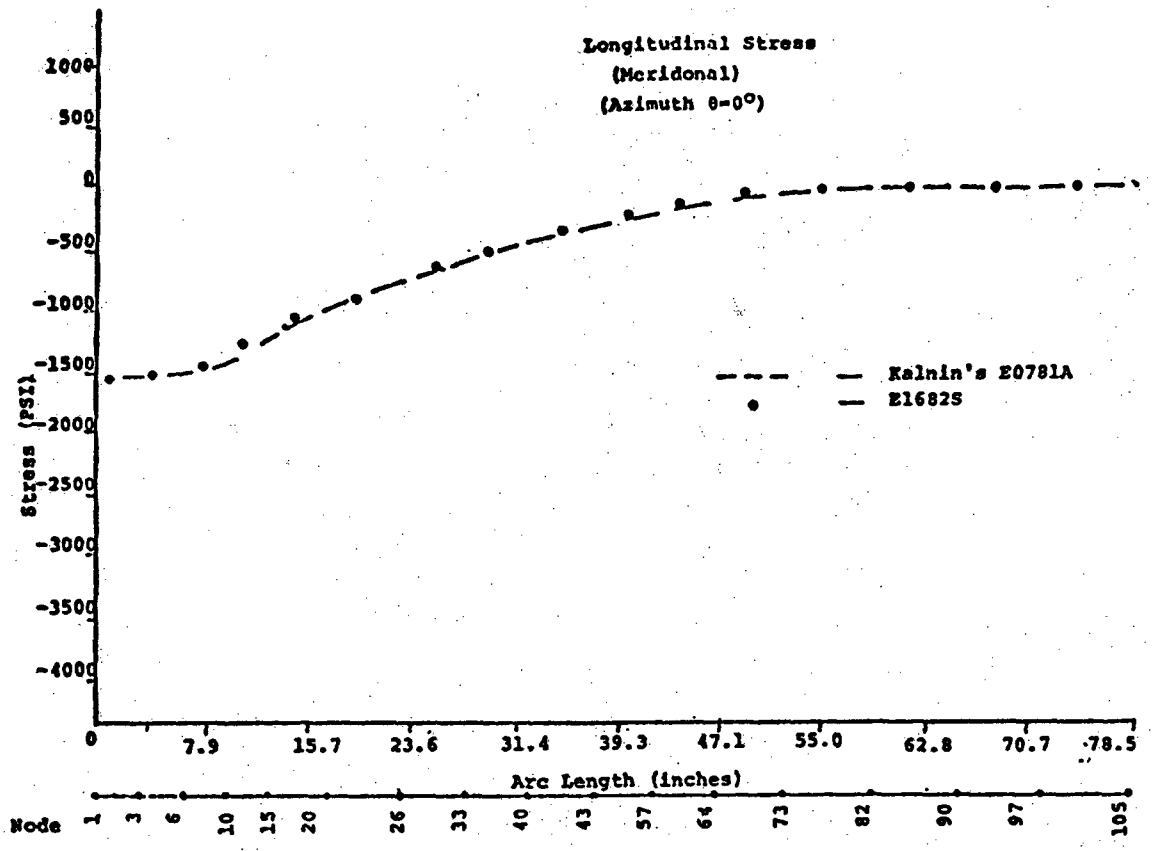
Longitudinal Stress
(Meridonal)
(Azimuth θ=0°)
1000
500
0
-500
-1000
-1500
-2000
-2500
-3000
-3500
-4000
Stress (PSI)
Kalnin's E0781A
E1682S
0
7.9
15.7
23.6
31.4
39.3
47.1
55.0
62.8
70.7
78.5
Arc Length (inches)
Node
1
3
6
10
15
20
26
33
40
43
57
64
73
82
90
97
105

Fig. A.30-4

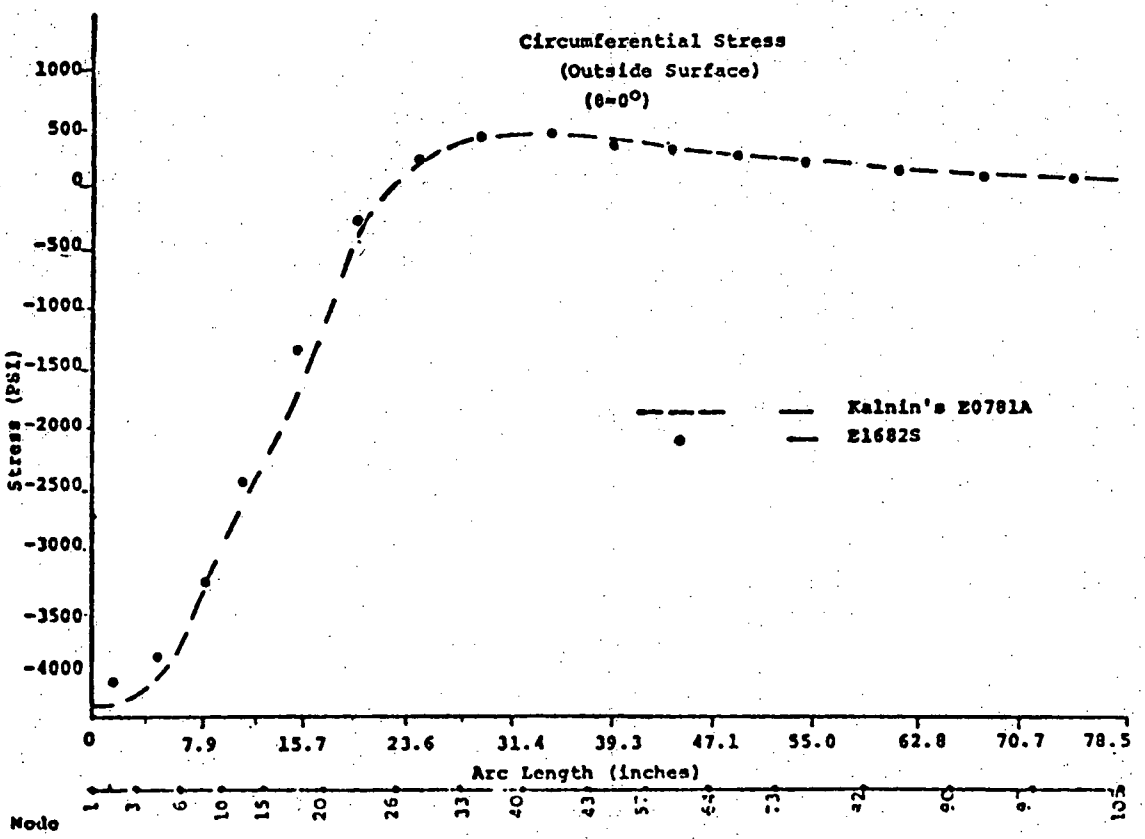
Circumferential Stress
(Outside Surface)
(θ=0°)
Stress (PSI)
1000
500
0
-500
-1000
-1500
-2000
-2500
-3000
-3500
-4000
Kalnin's E0781A
E1682S
0
7.9
15.7
23.6
31.4
39.3
47.1
55.0
62.8
70.7
78.5
Arc Length (inches)
Node

Fig. A.30-5

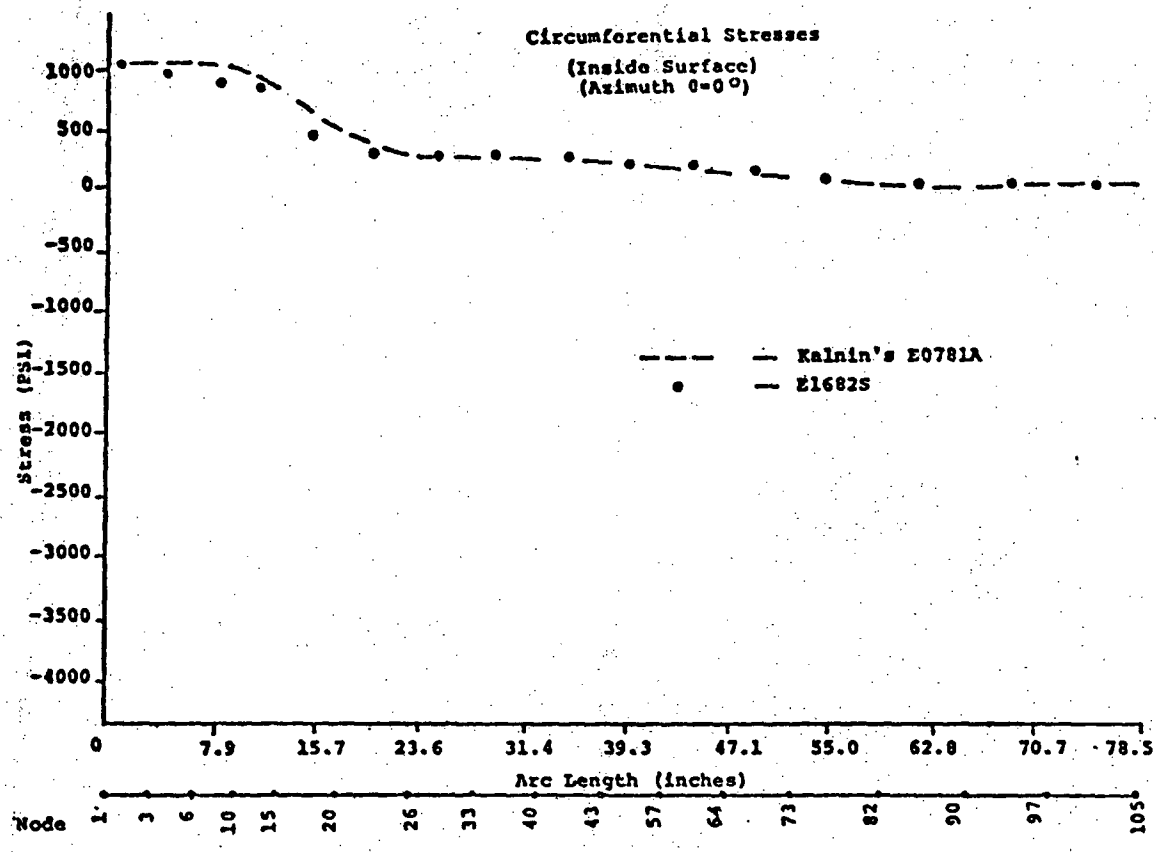
Circumferential Stresses
(Inside Surface)
(Azimuth θ=0°)
1000
500
0
-500
-1000
-1500
-2000
-2500
-3000
-3500
-4000
Stress (PSI)
Kalnin's E0781A
E1682S
0
7.9
15.7
23.6
31.4
39.3
47.1
55.0
62.8
70.7
78.5
Arc Length (inches)
Node
1
3
6
10
15
20
26
33
40
43
57
64
73
82
90
97
105

Fig. A.30-6

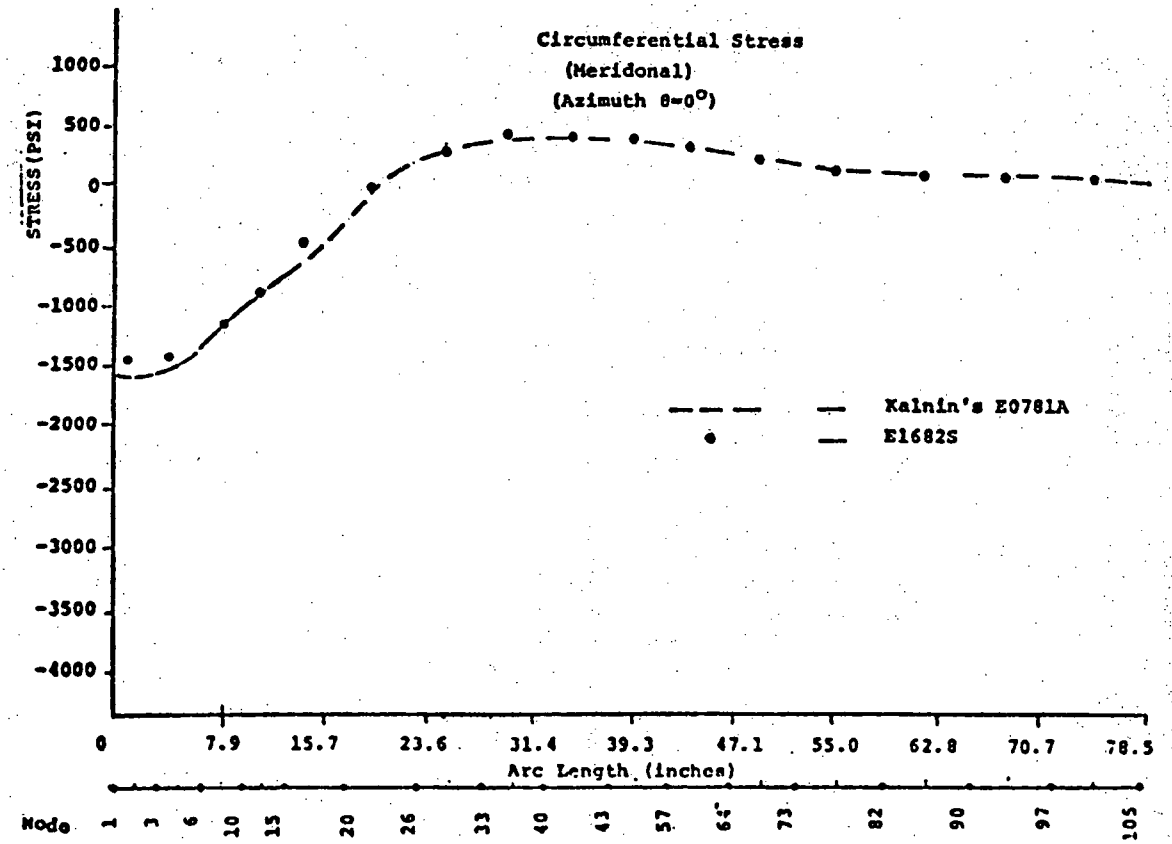
Circumferential Stress
(Meridonal)
(Azimuth θ=0°)
STRESS (PSI)
1000
500
0
-500
-1000
-1500
-2000
-2500
-3000
-3500
-4000
Kalnin's E0781A
E1682S
0
7.9
15.7
23.6
31.4
39.3
47.1
55.0
62.8
70.7
78.5
Arc Length (inches)
Node
1
3
6
10
15
20
26
33
40
43
57
65
73
82
90
97
105

Fig. A.30-7

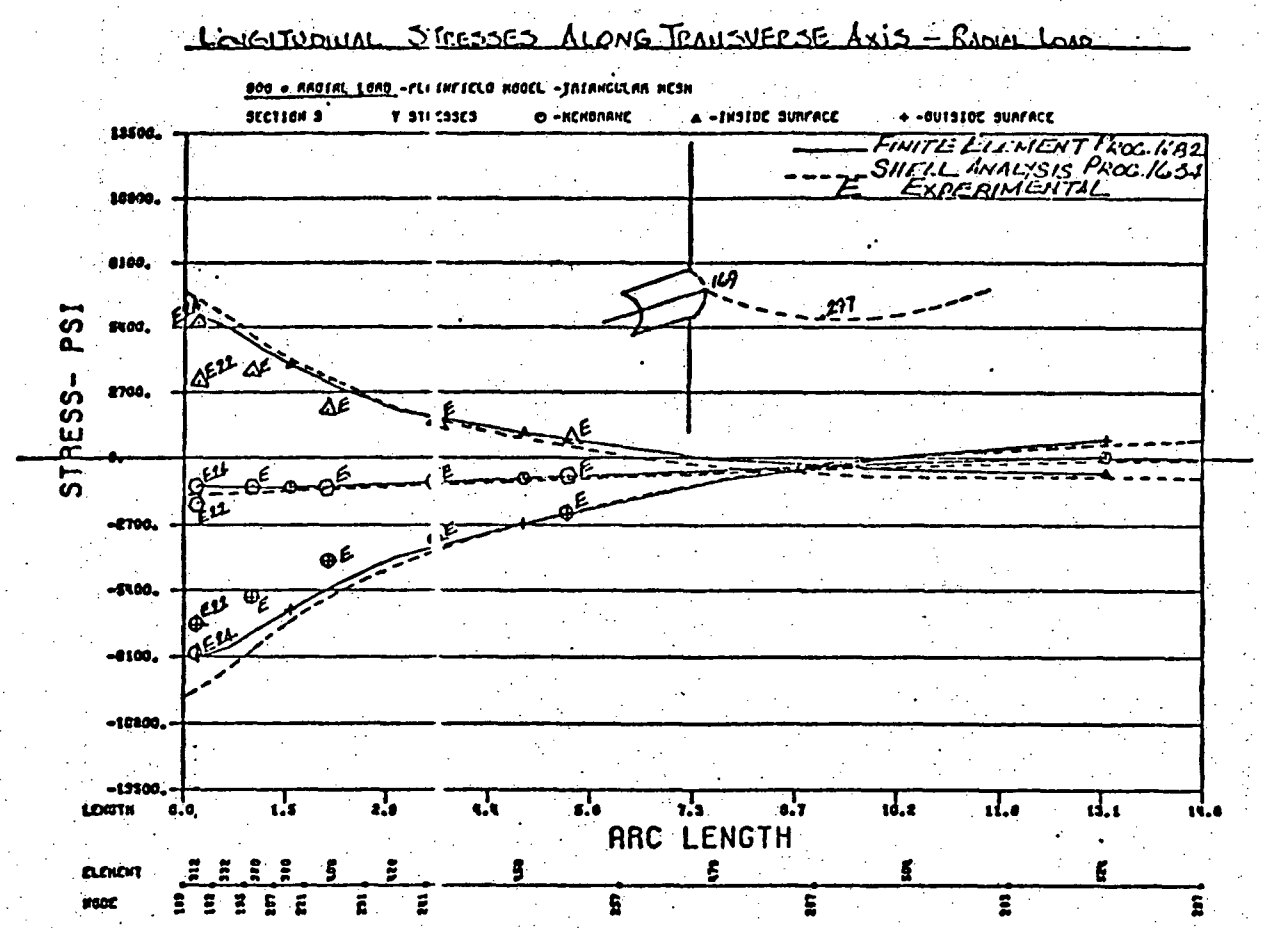
LONGITUDINAL STRESSES ALONG TRANSVERSE AXIS - RADIAL LOAD
SECTION 3
O -MEMBRANE
▲ -INSIDE SURFACE
+ -OUTSIDE SURFACE
FINITE ELEMENT PROC. 1482
SHELL ANALYSIS PROC. 1634
E EXPERIMENTAL
STRESS- PSI
ARC LENGTH
LENGTH
ELEMENT
NODE

Fig. A.30-8

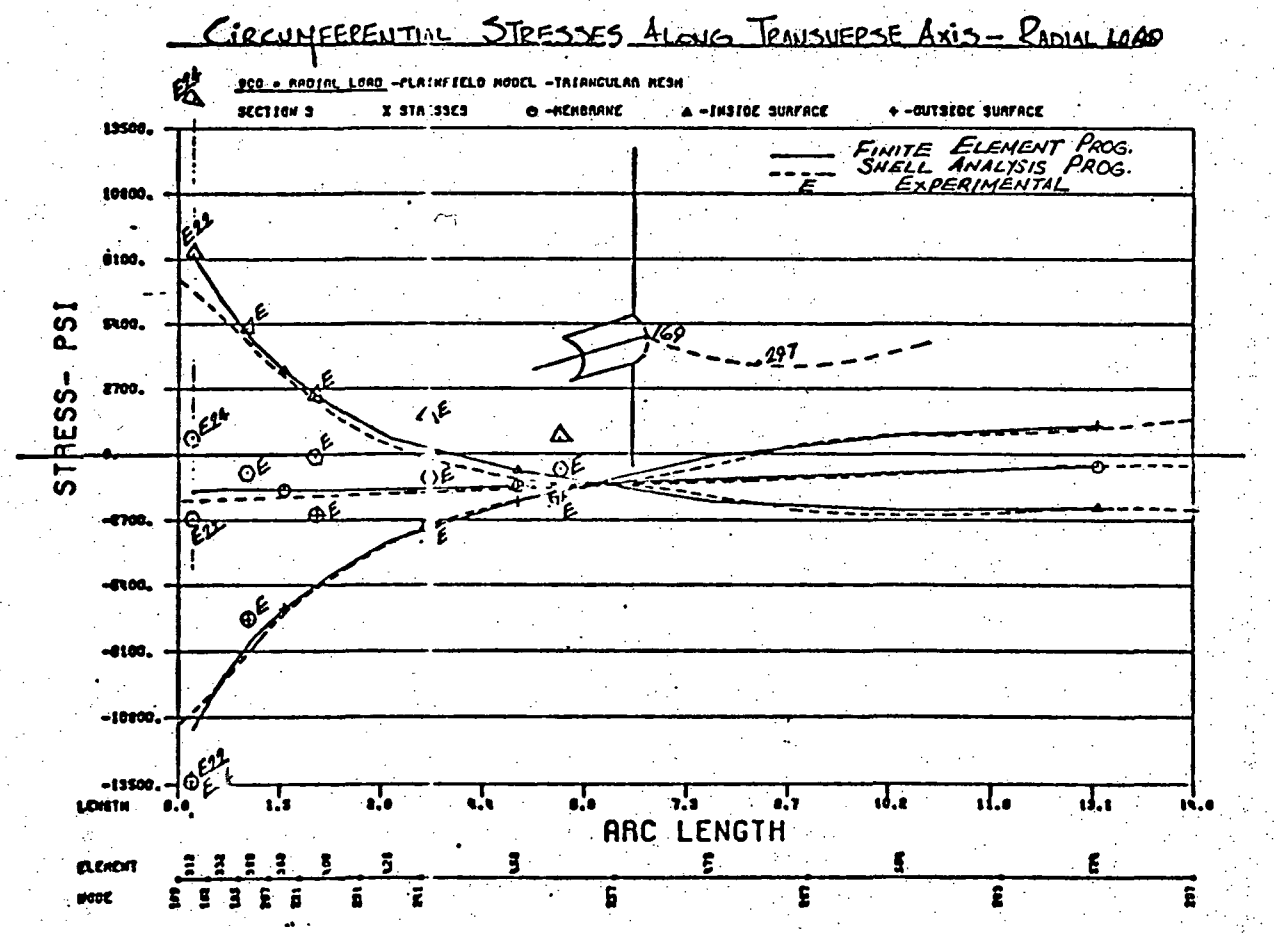

Fig. A.30-9

Amend. 45
July 1978

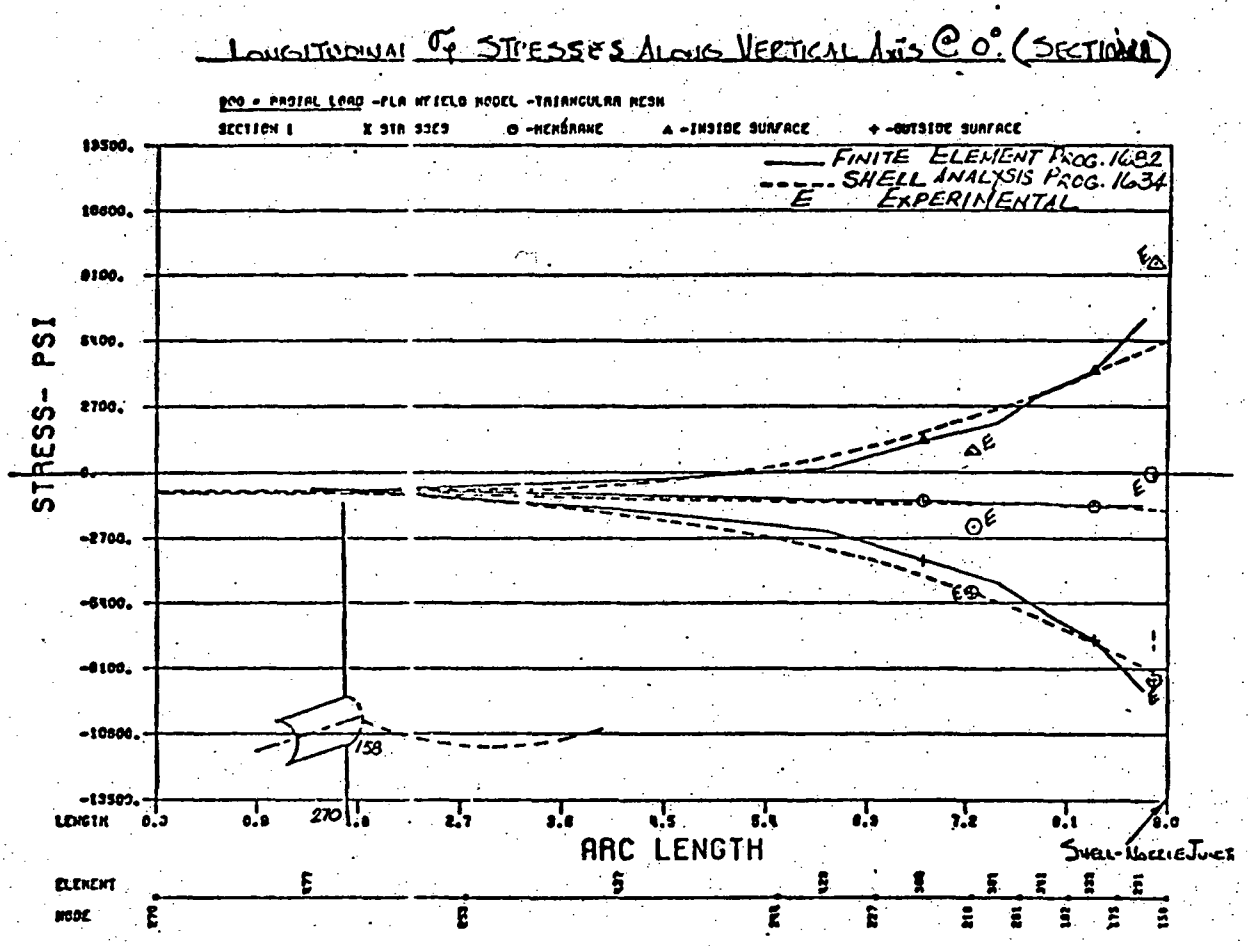
LONGITUDINAL σφ STRESSES ALONG VERTICAL AXIS @ 0° (SECTION A)
FINITE ELEMENT PROG. 1682
SHELL ANALYSIS PROG. 1634
E EXPERIMENTAL
STRESS- PSI
ARC LENGTH
LENGTH
ELEMENT
NODE
158
270

Fig.A.30-10

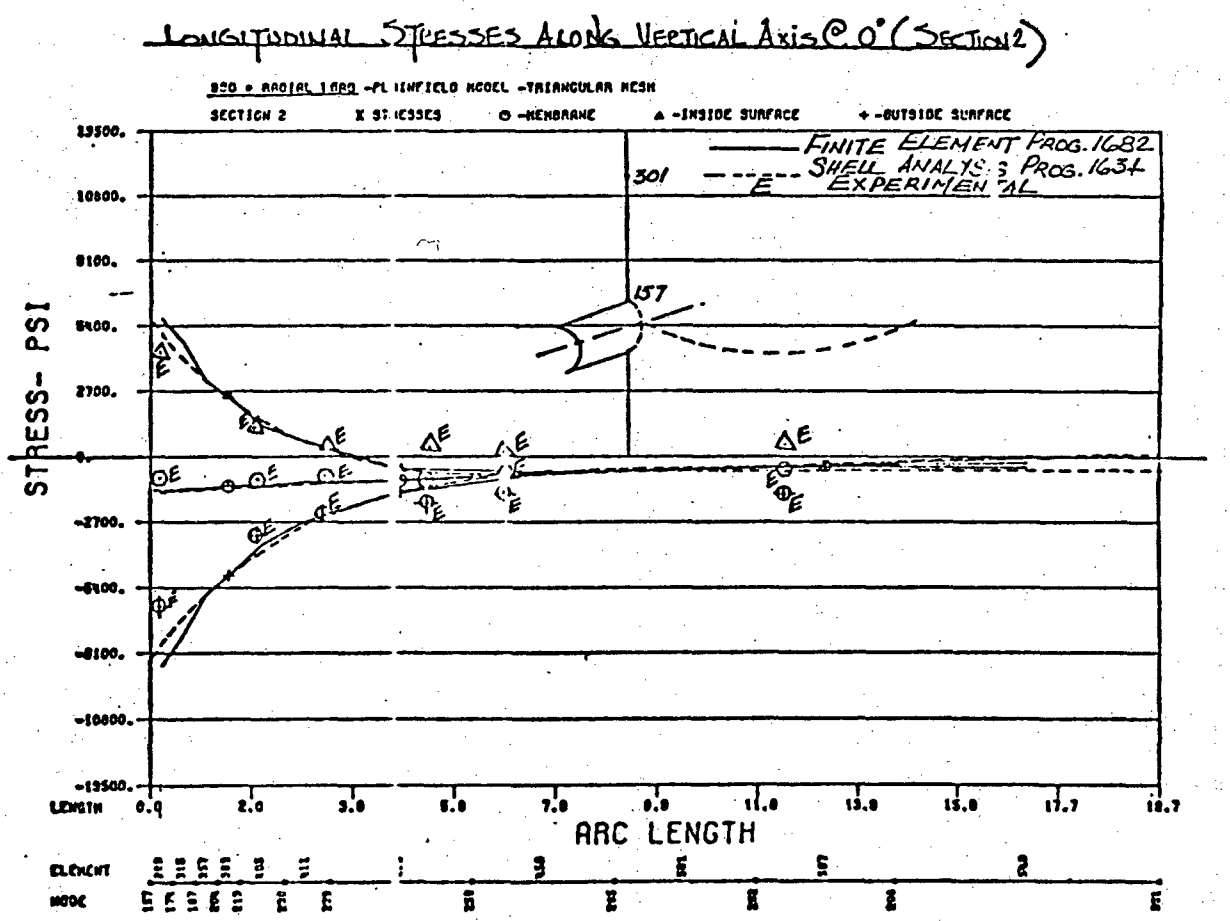

Fig.A.30-11

Amend. 45
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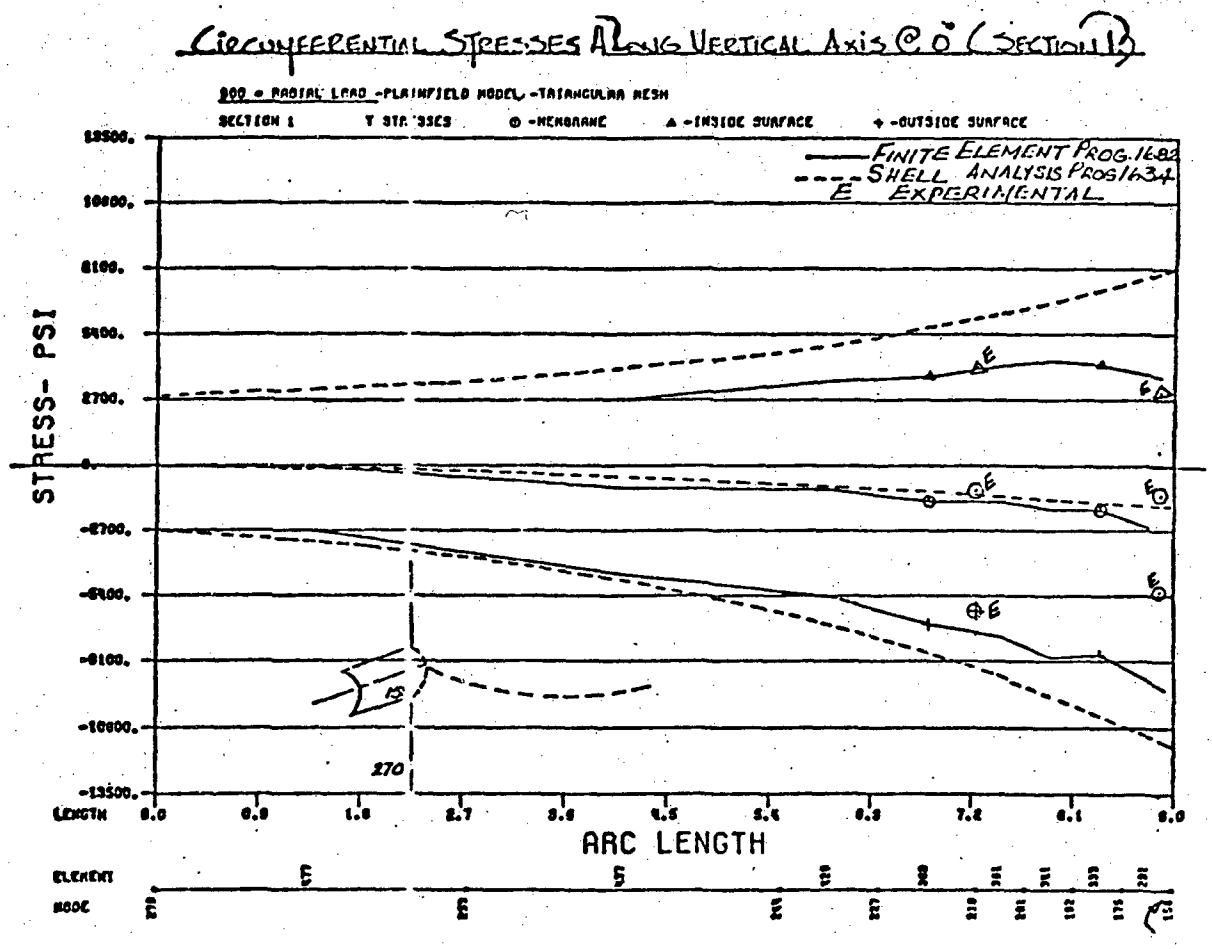
Circumferential Stresses Along Vertical Axis @ 0° (Section 1)
Finite Element Prog. 1682
Shell Analysis Prog 1634
E Experimental
STRESS- PSI
ARC LENGTH
270

Fig.A.30-12

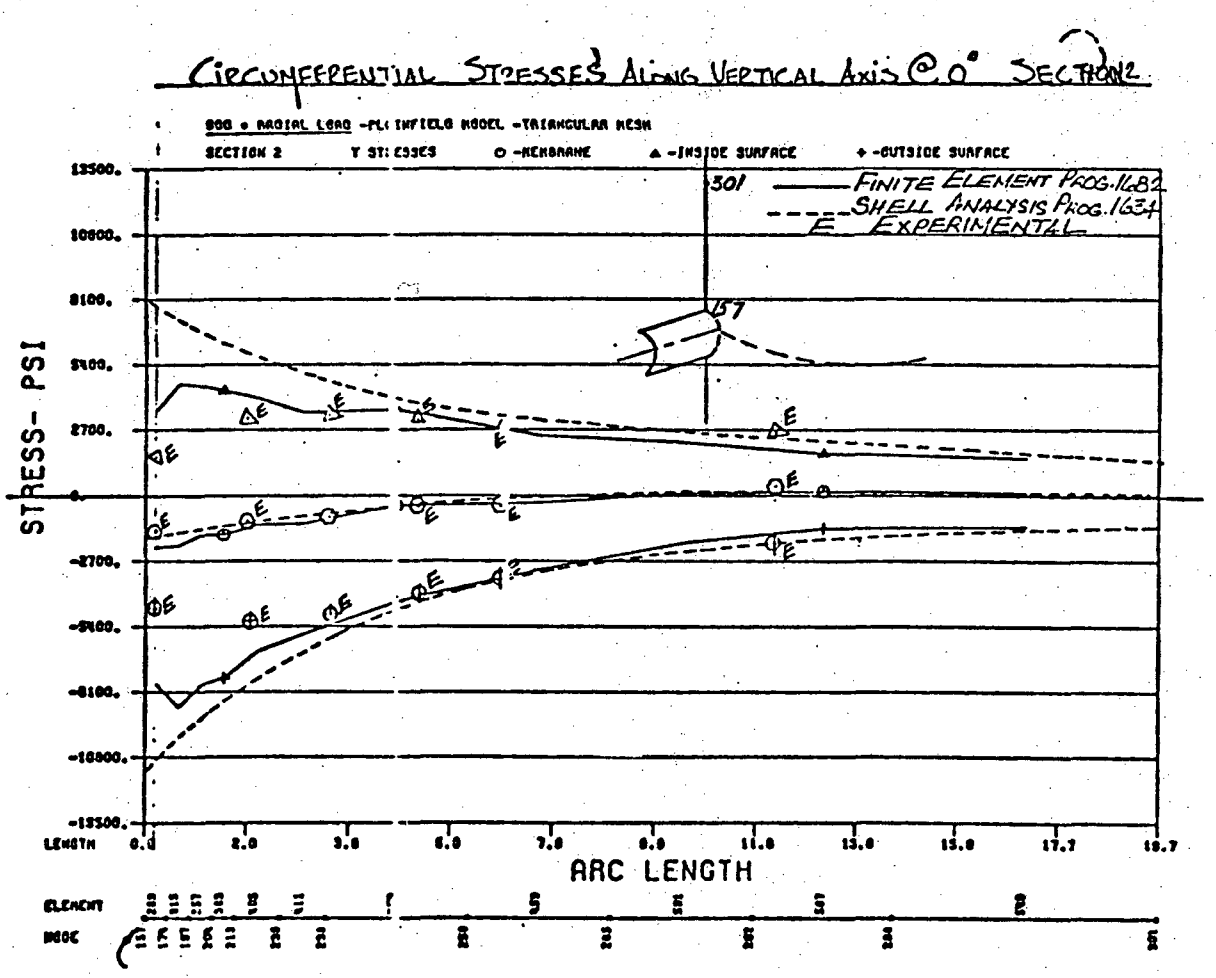

Fig.A.30-13

Amend. 45
July 1978

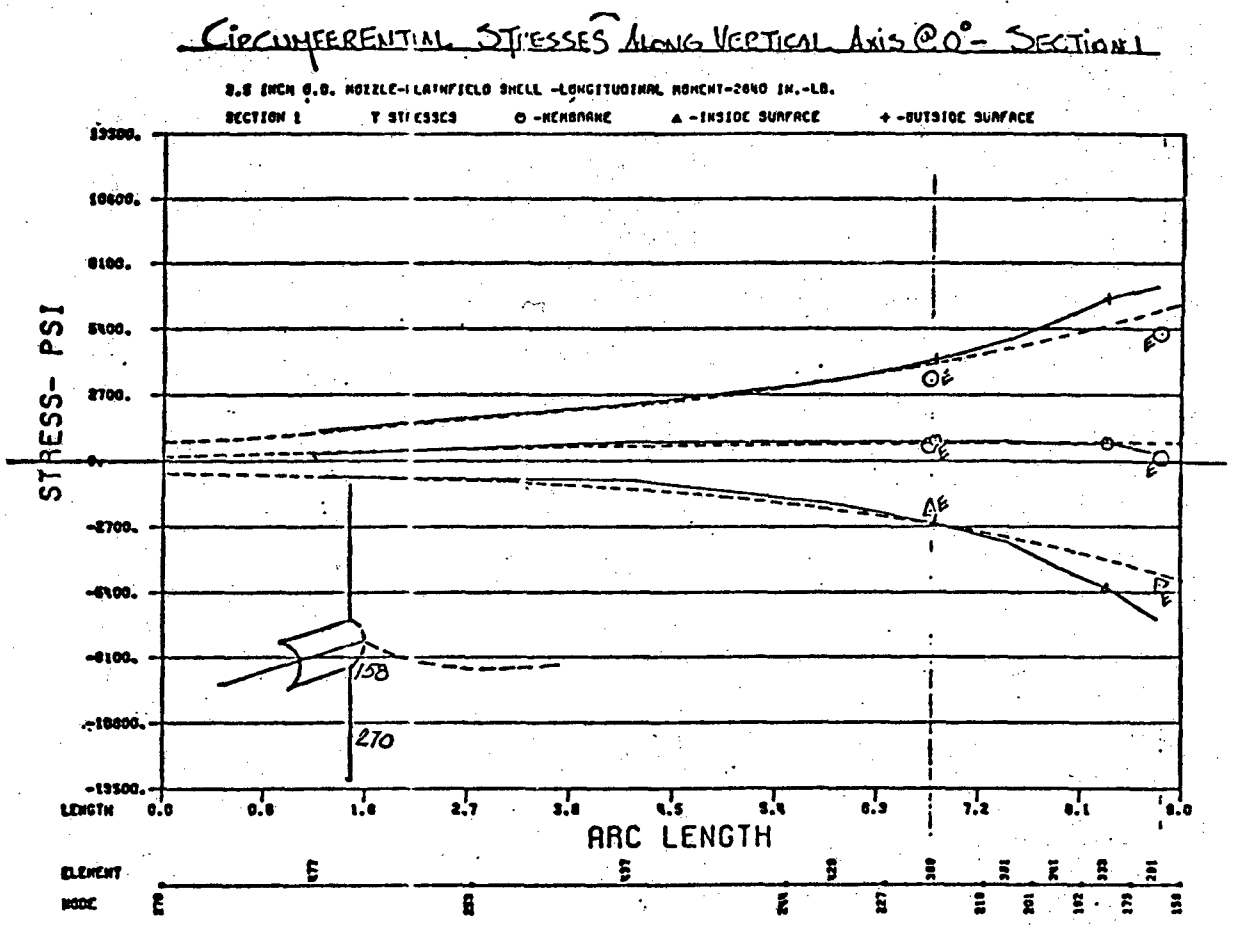
CIRCUMFERENTIAL STRESSES ALONG VERTICAL AXIS @ 0° - SECTION 1
8.8 INCH O.D. NOZZLE-FAIRFIELD SHELL -LONGITUDINAL MOMENT=2040 IN.-LB.
SECTION 1
T STRESSES
O -MEMBRANE
Δ -INSIDE SURFACE
+ -OUTSIDE SURFACE
STRESS- PSI
13300.
10600.
8100.
5400.
2700.
0.
-2700.
-5400.
-8100.
-10800.
-13500.
158
270
LENGTH
0.0
0.9
1.8
2.7
3.6
4.5
5.4
6.3
7.2
8.1
9.0
ARC LENGTH
ELEMENT
NODE

Fig.A.30-14

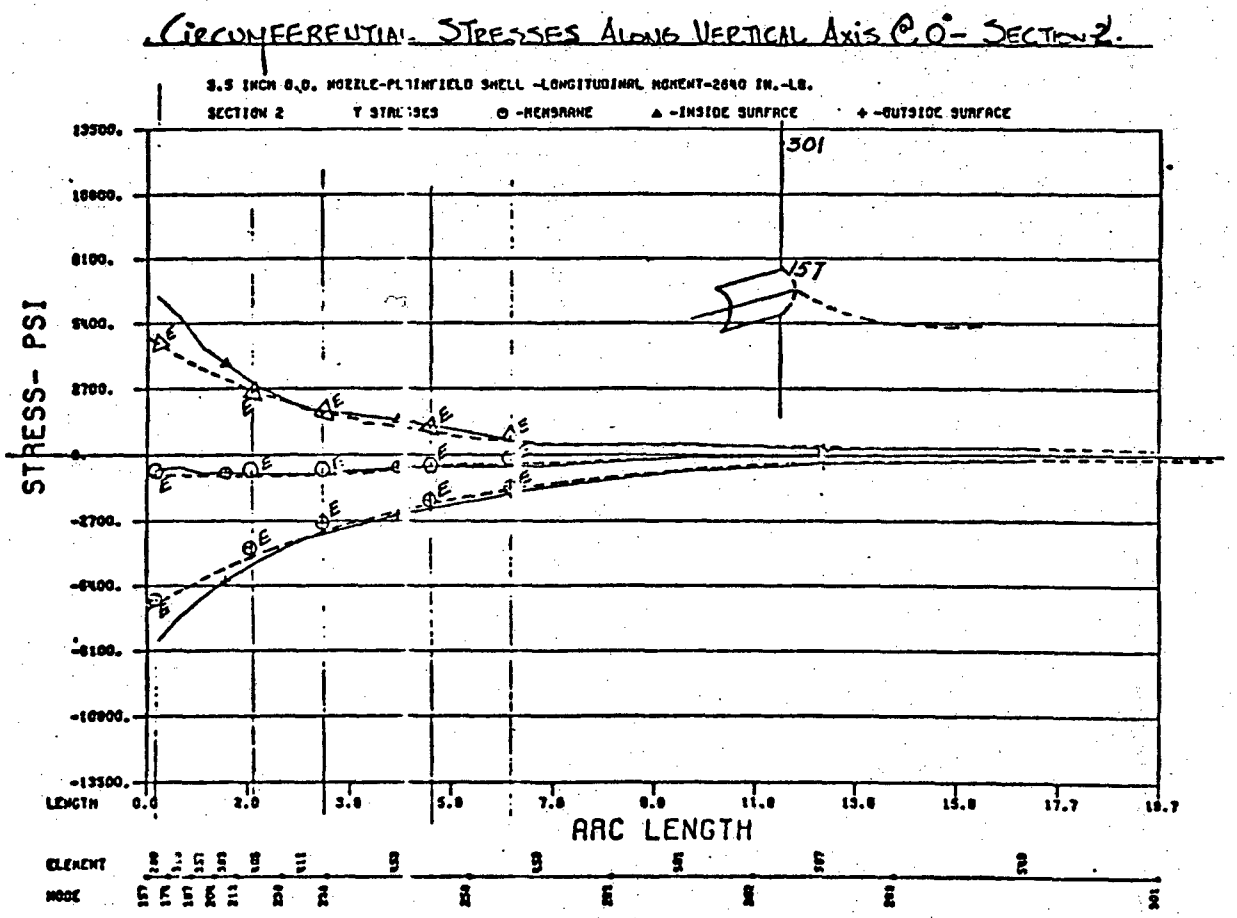
Circumferential Stresses Along Vertical Axis @ 0° - Section 2.
SECTION 2
⊖ -MEMBRANE
▲ -INSIDE SURFACE
+ -OUTSIDE SURFACE
301
157
STRESS- PSI
LENGTH
ARC LENGTH
ELEMENT
NODE

Fig.A.30-15

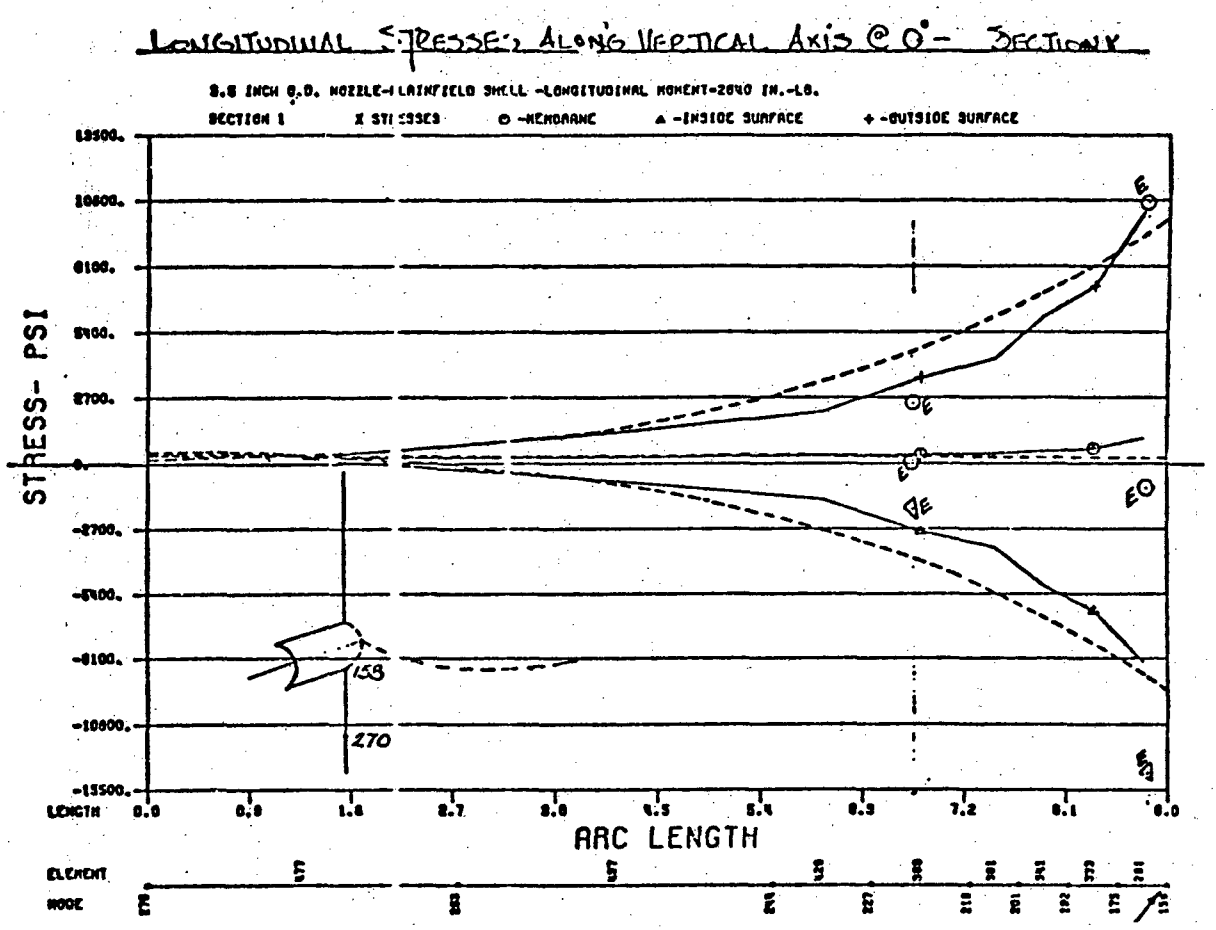

Fig.A.30-16

Amend. 45
July 1978

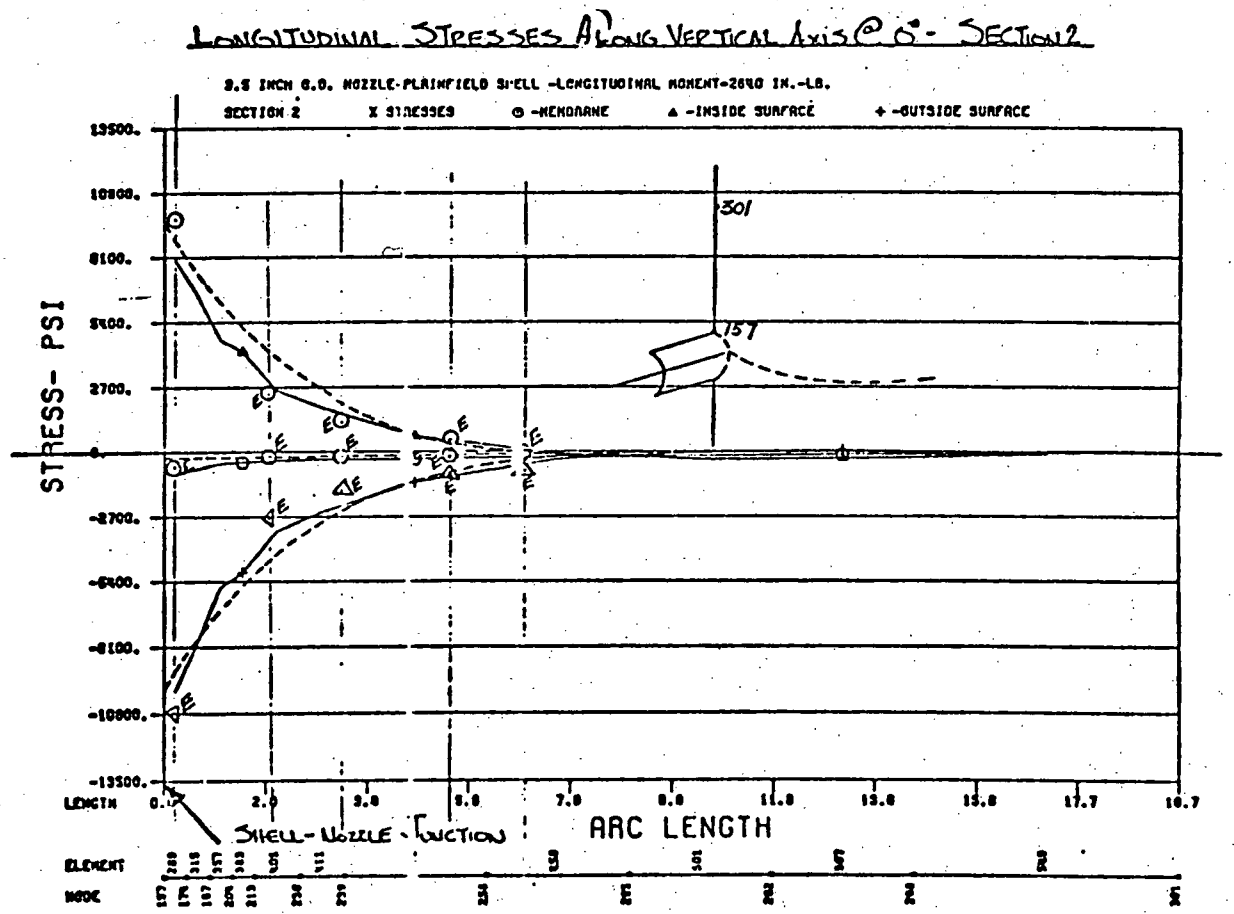
LONGITUDINAL STRESSES ALONG VERTICAL AXIS @ 0° - SECTION 2
SECTION 2
X STRESSES
⊙ -MEMBRANE
▲ -INSIDE SURFACE
+ -OUTSIDE SURFACE
STRESS- PSI
13500.
10800.
8100.
5400.
2700.
0.
-2700.
-5400.
-8100.
-10800.
-13500.
LENGTH
ARC LENGTH
SHELL-NOZZLE-JUNCTION
301
157
ELEMENT
NODE

Fig.A.30-17

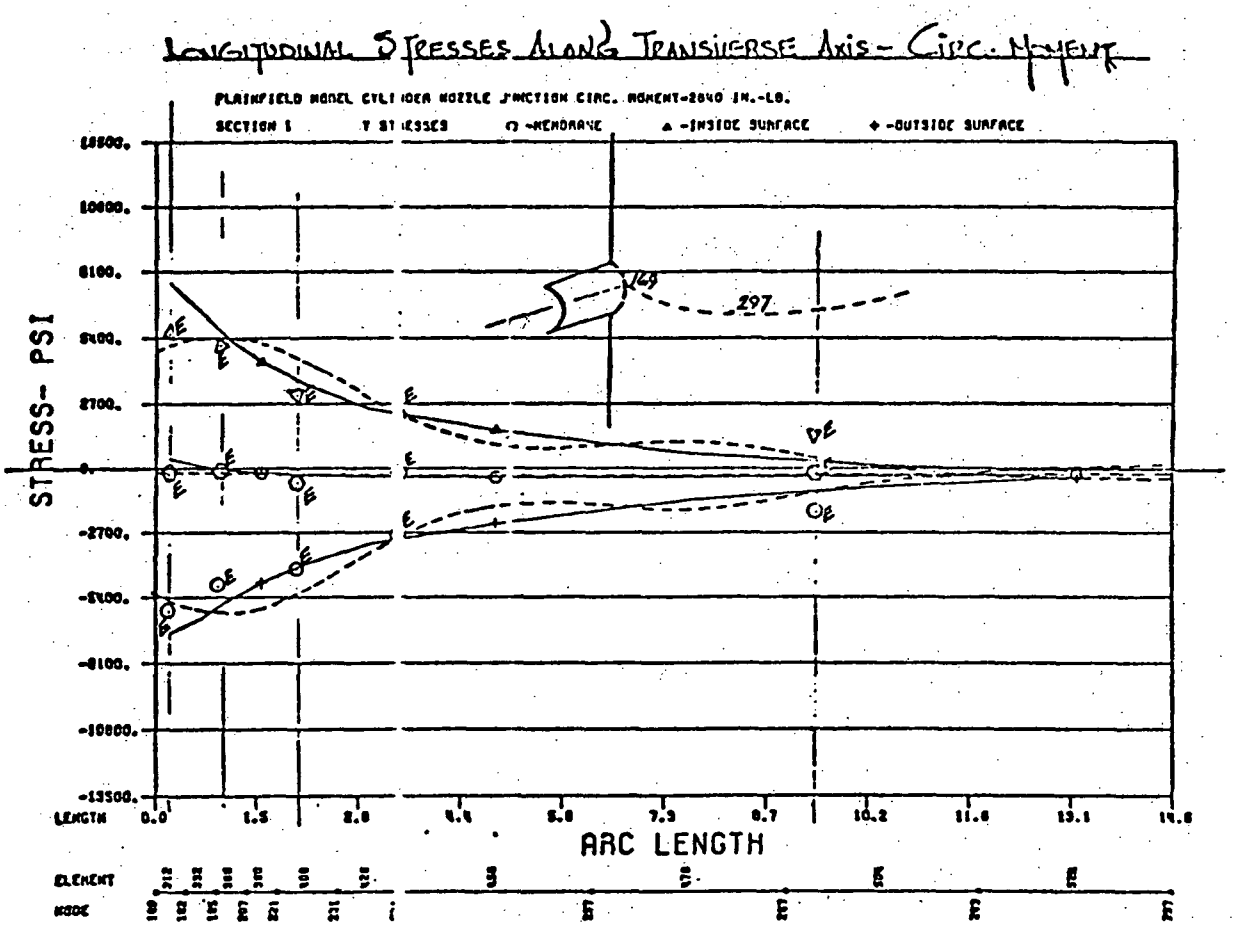

Fig.A.30-18

Amend. 45
July 1978

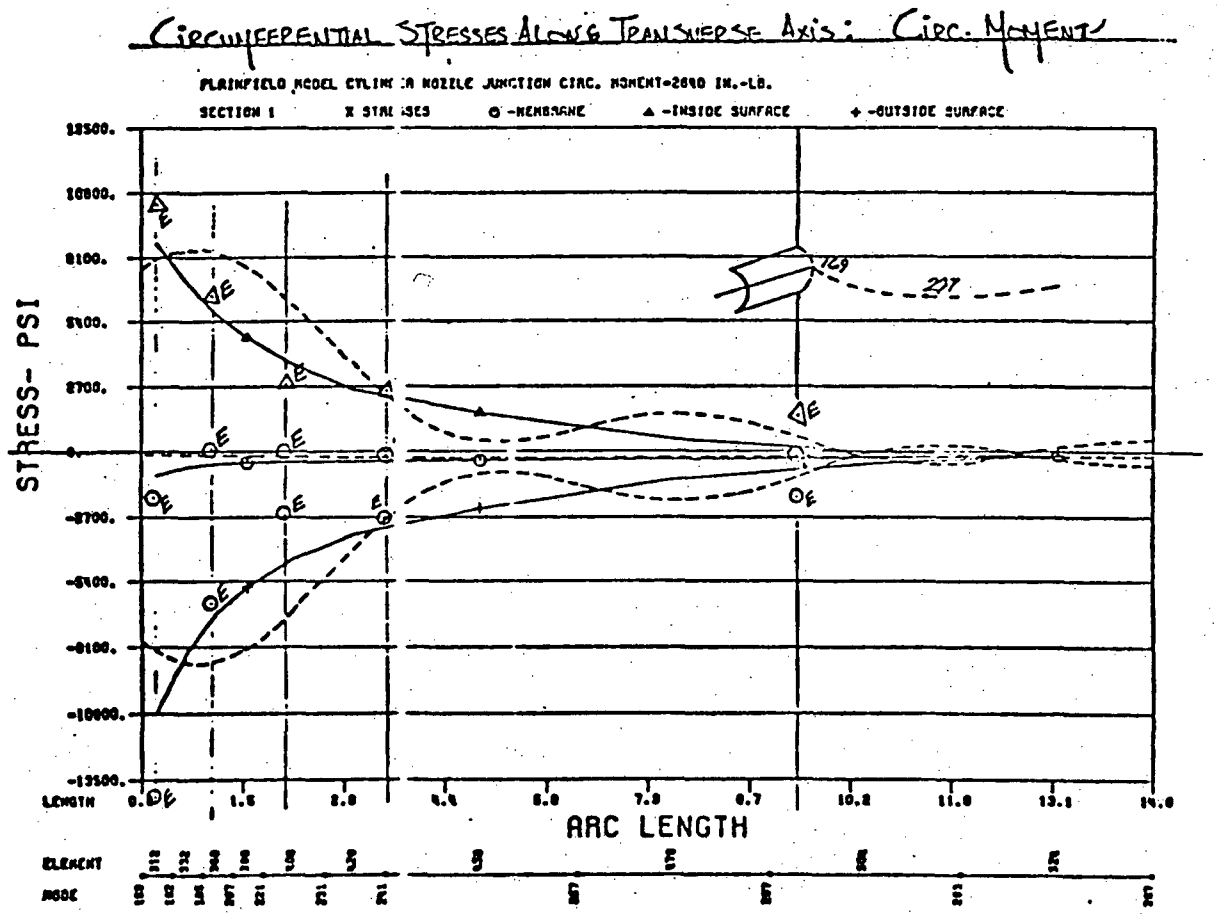

Fig.A.30-19

Amend. 45
July 1978

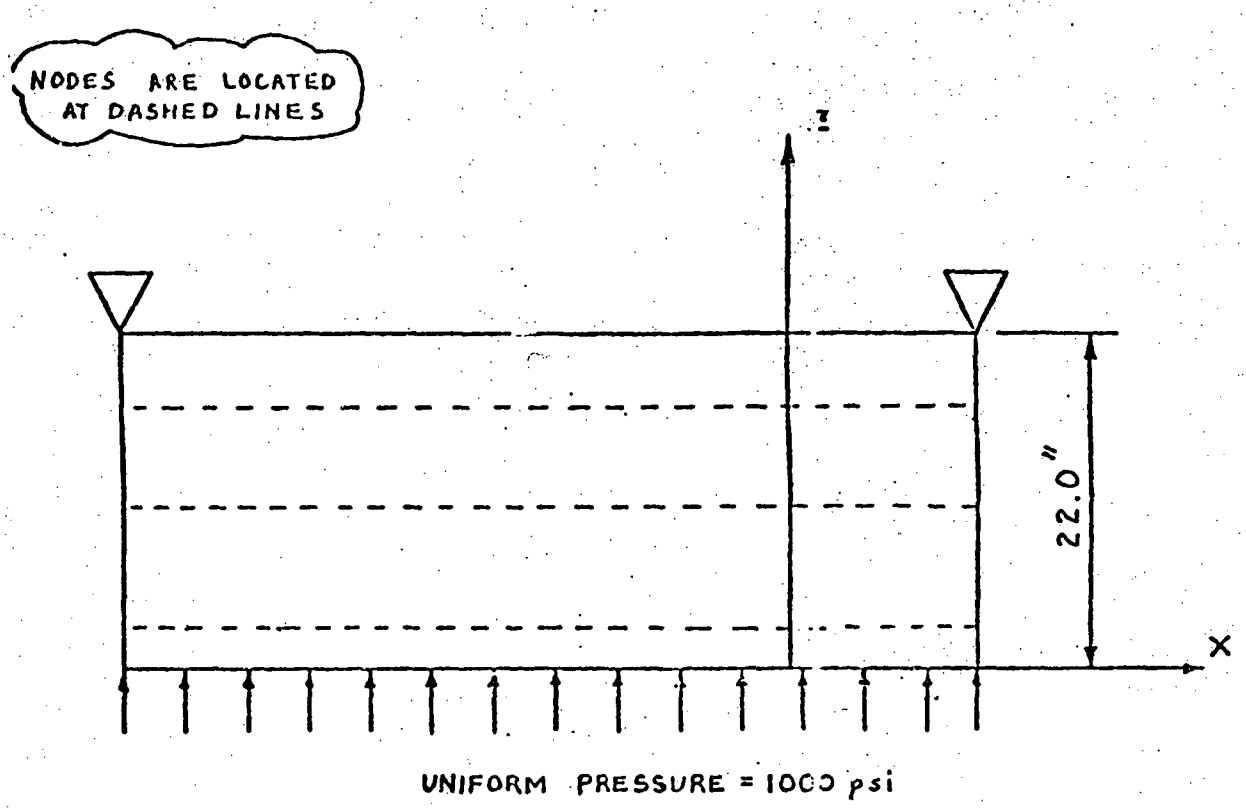

FIGURE A.30-20

SIMPLY SUPPORTED CIRCULAR PLATE

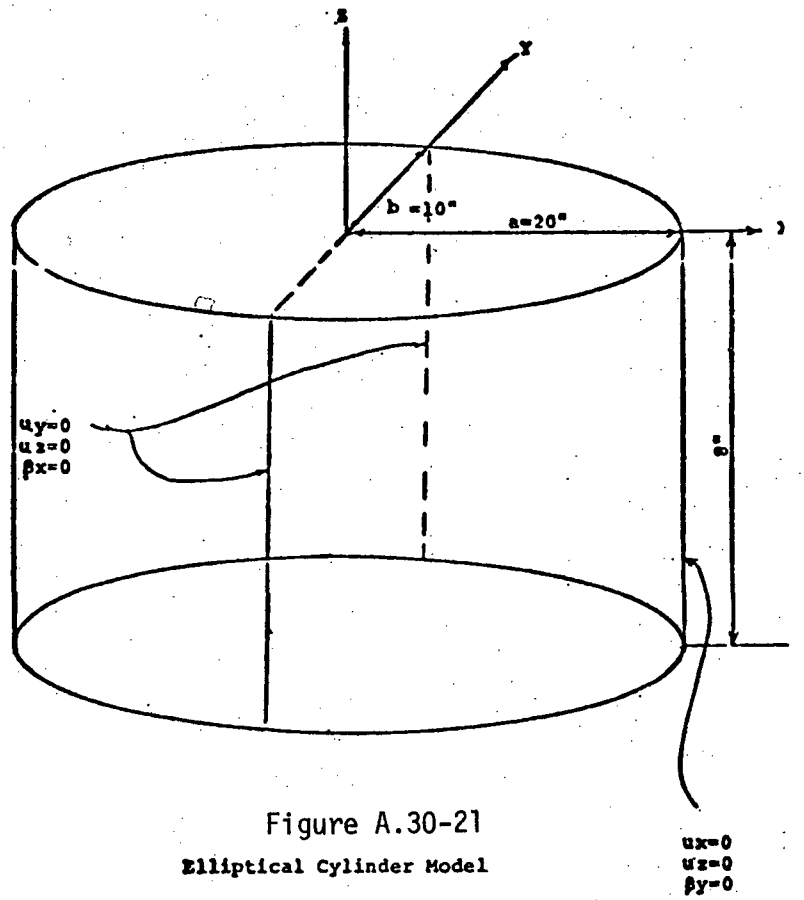

Figure A.30-21

Elliptical Cylinder Model

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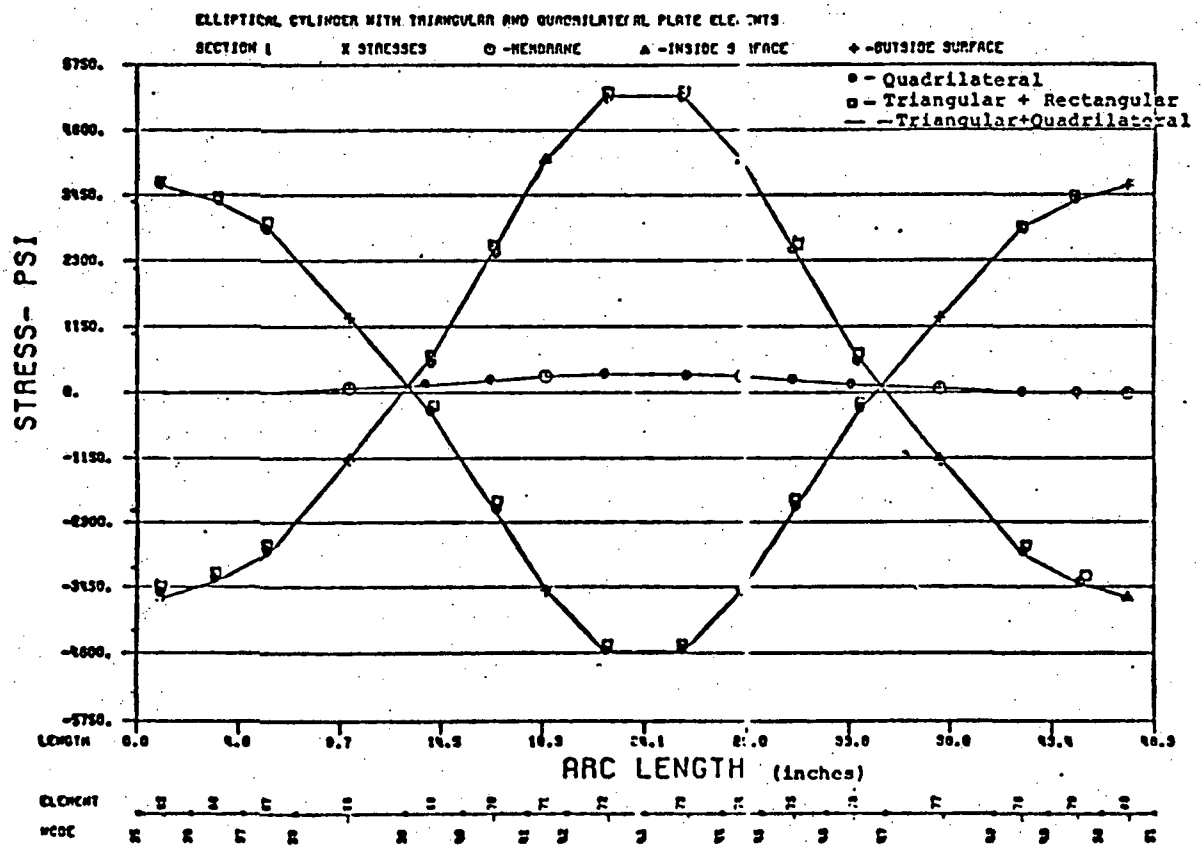

Figure A.30-22 X-DIRECTION STRESSES

Amend. 45
July 1978

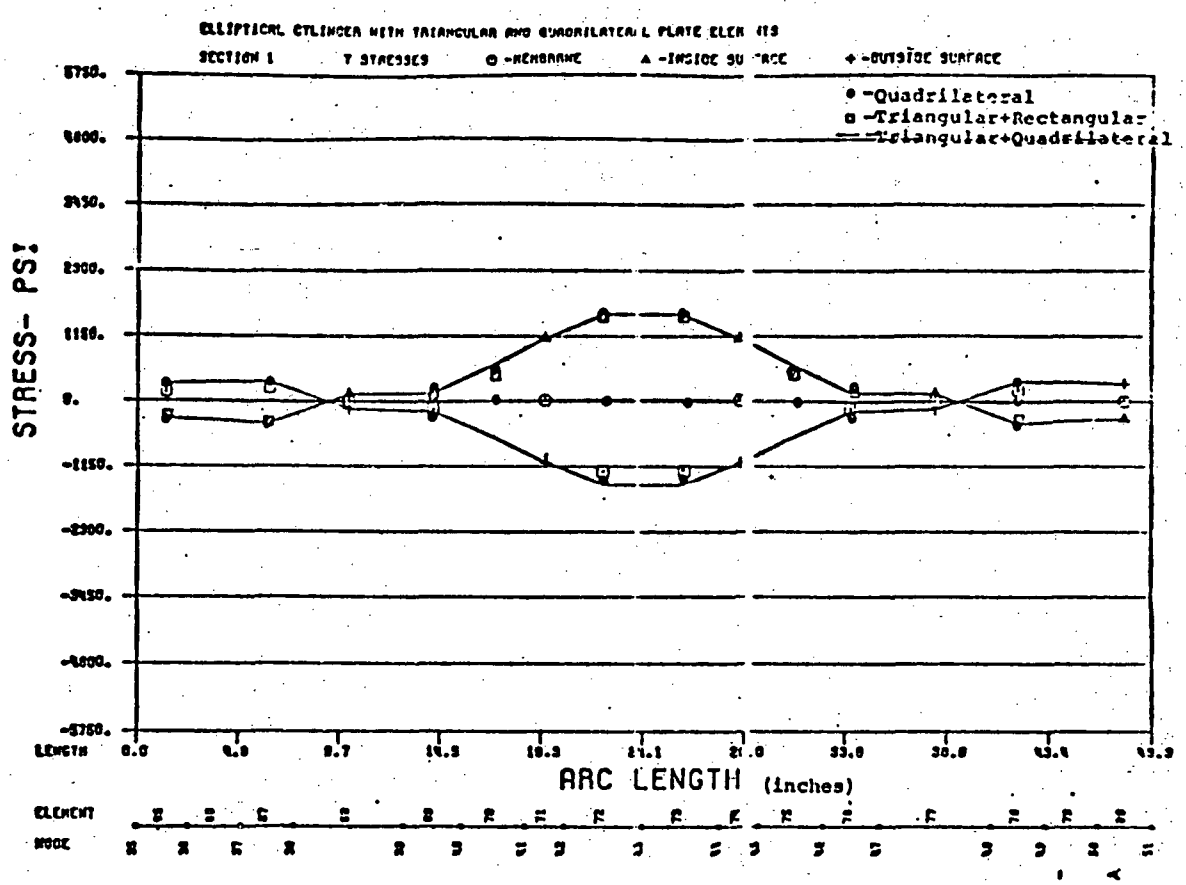

Figure A.30-23 Y DIRECTION STRESSES.

Amend. 45
July 1978

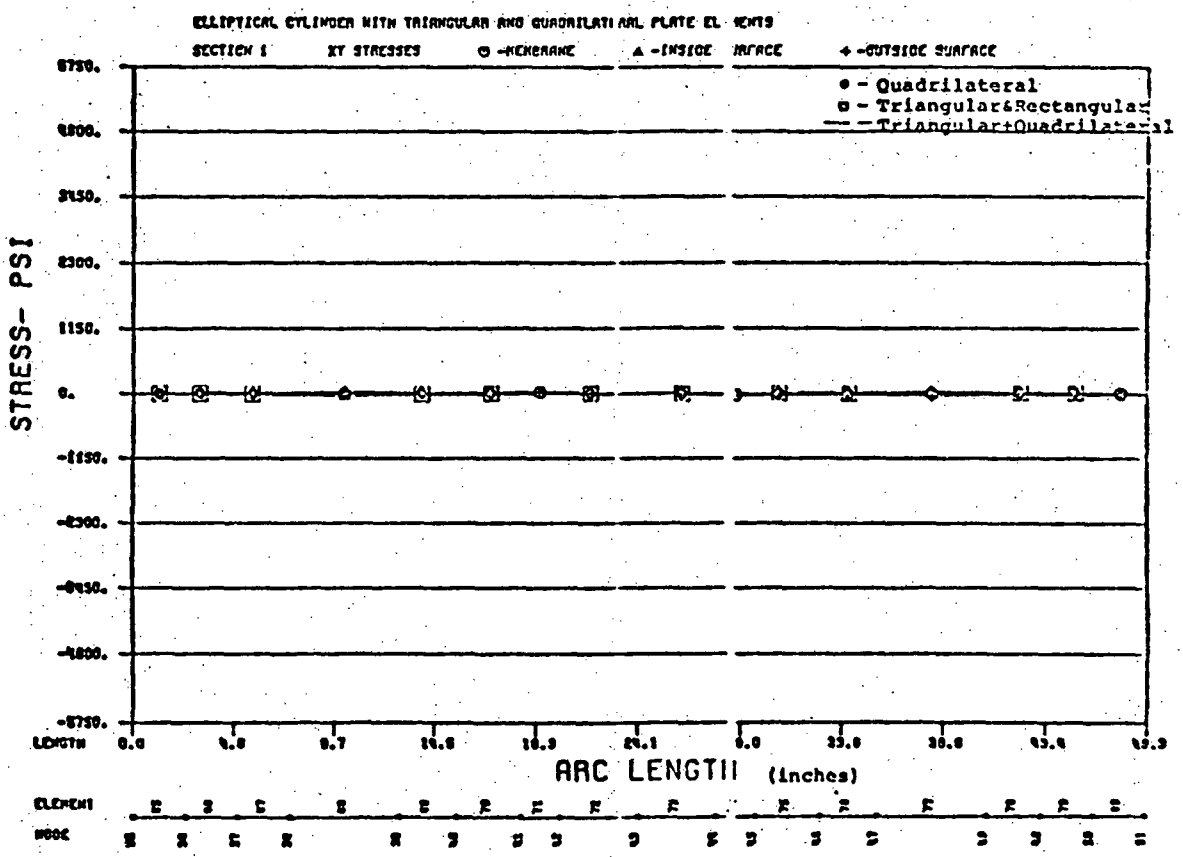

Figure A.30-24 XY STRESSES.

Amend. 45
July 1978

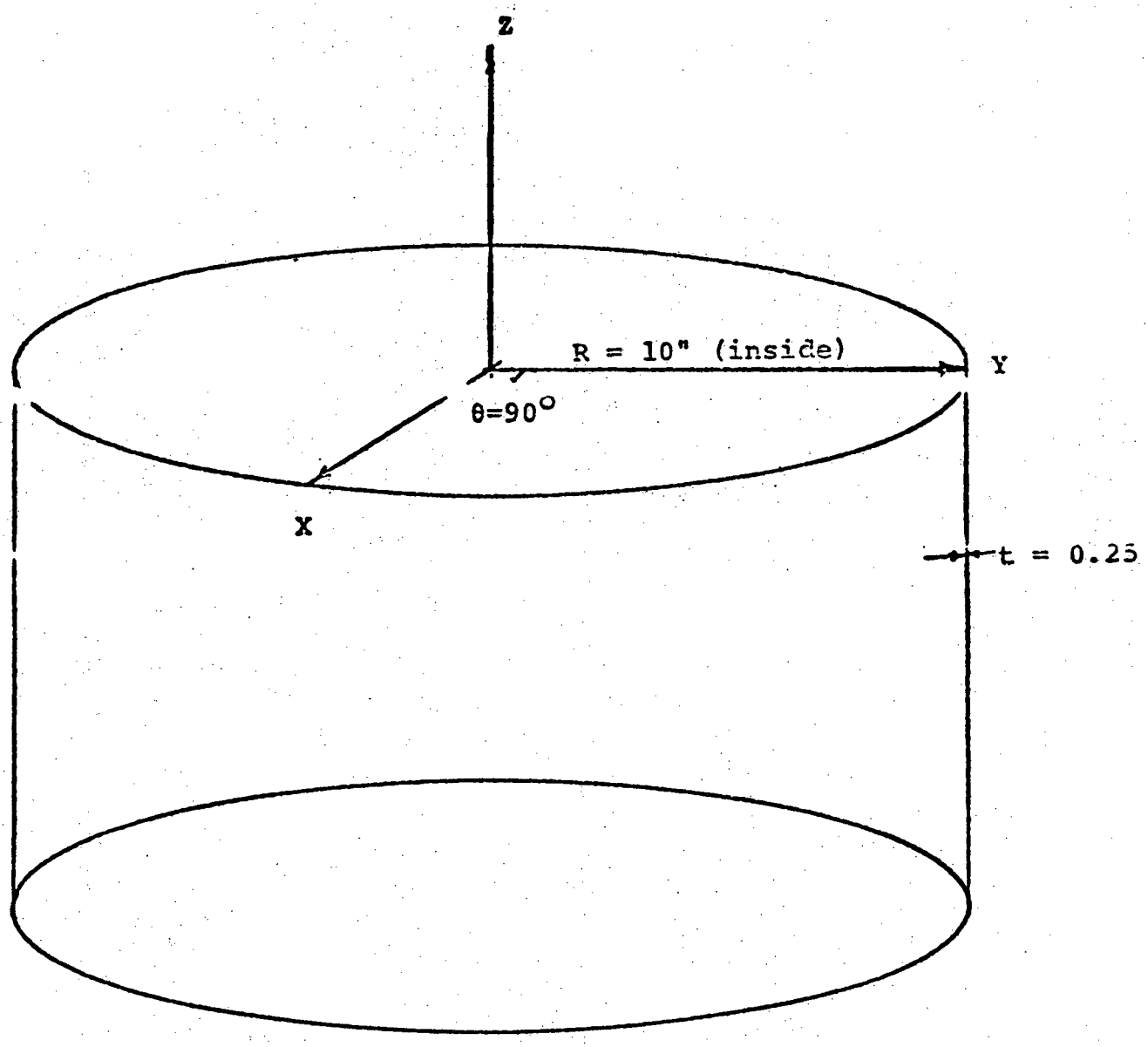

Figure A.30-25

RIGHT CIRCULAR CYLINDER

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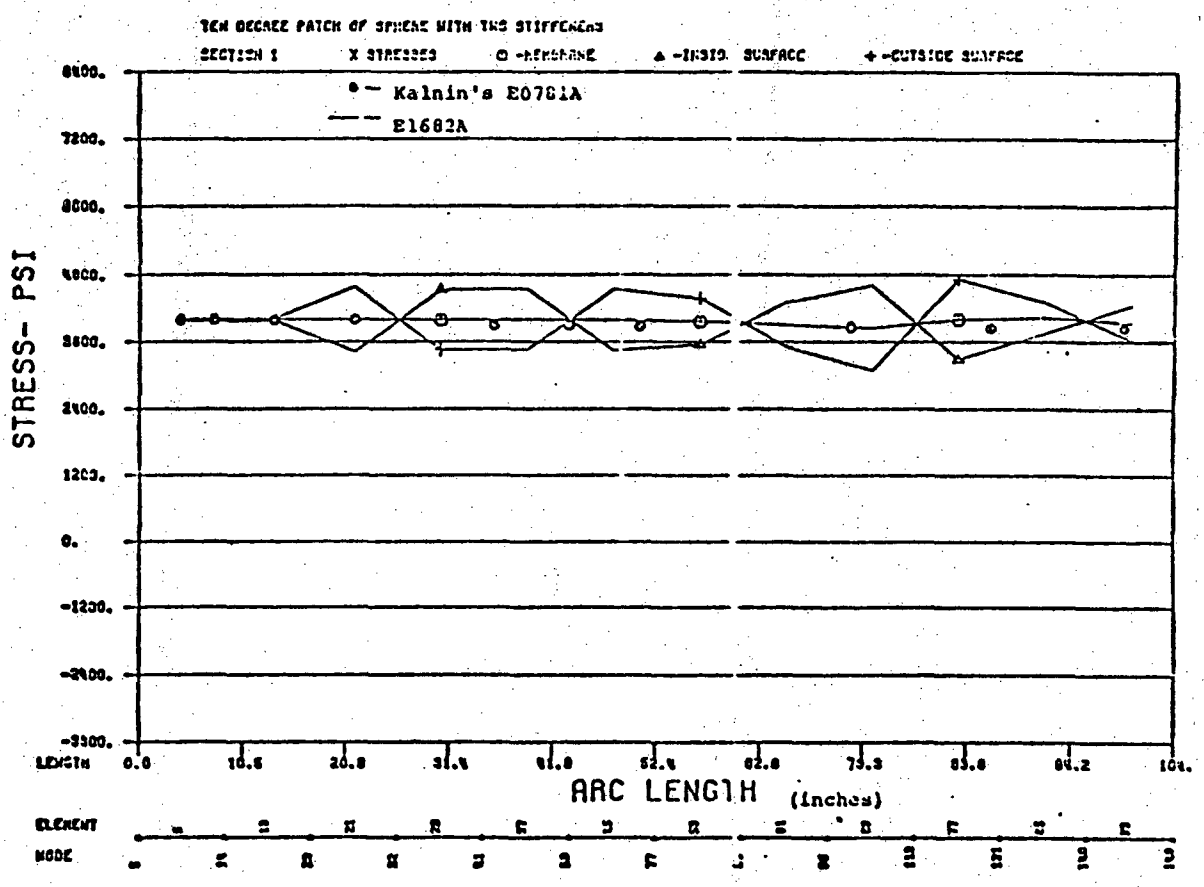

Figure A.30-26 Y DIRECTIONAL STRESSES

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A.31

E1755A-Nonlinear Temperature Analysis of 3-D Solids

This program is used for analysis of one dimensional steady state conduction at assigned temperature nodes and zero temperature nodes.

Availability

This program is available on the IBM 370, Model 165 computer of the Chicago Bridge and Iron Company.

Verification

Verification of this program is achieved through the use of the following problem.

The structure is a portion of a thick wall with a large temperature gradient across it caused by having a surface at 0°F and the opposite side at 800°F.

The results obtained with program E1755A are tabulated in Table A.31-1 and compared to the exact solution show good agreement.

Application

Program E1755A will be used in the design of the closure head assembly.

Amend. 45
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Table A.31-1

COMPARISON OF RESULTS

ELEVATION Z IN.	TEMPERATURE THEORY	TEMPERATURE E1755A	E1755A % Error
1.0	0.0	0.0	0.0
2.0	100.0	99.7	0.3
3.0	200.0	199.5	0.25
4.0	300.0	299.4	0.2
5.0	400.0	399.4	0.15
6.0	500.0	499.4	0.12
7.0	600.0	599.5	0.083
8.0	700.0	699.7	0.043
9.0	800.0	800.0	0.0

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A.32 E1739A-Form Factors for Exchange of Radiant Energy

This program calculates radiation form factors for axisymmetrical sections of cylinders radiating internally to annular and circular sections of their bases.

Availability

This program is available on the IBM 370, Model 165 Computer of the Chicago Bridge and Iron Company.

Verification

Verification of this program is achieved through the use of the following problems:

1. Form Factors for a Cylinder

   The problem consists of 9 nodes which form a cylindrical enclosure. This cylinder has 3 surfaces at the top, 2 at the bottom and 3 along the side. The problem dimensions are shown in Figure A.32-1.

   The form factors generated are checked by hand calculations and are given in Table A.32-1. The rows are totalled to check the accuracy of the program solution.

2. Form Factors Between Two Cylindrical Surfaces

   The problem consists of an axisymmetric two inch enclosure between cylindrical surfaces 2 and 6, see Figure A.32-2. This enclosure has 1 surface at the bottom and 3 along the top. Seven cases are run for this problem with the radius varying for each case as shown in Figure A.32-2.

   Form factor results are given in Table A.32-2. Form factor results are plotted in Figure A.32-3.

Application

Program E1739A will be used in the design of the closure head assembly.

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TABLE A.32-1

E1739A FORM REACTOR RESULTS

TO SURFACE *

FORM FACTOR FROM SURFACE	1	2	3	4	5	6	7	8	TOTAL
1	0	0	.3802	.2949	.1334	.0948	.0672	.0244	.9999
2	0	0	.5608	.1836	.0960	.0837	.0561	.0197	.9999
3	.1267	.2336	.2798	.1857	.0886	.0400	.0333	.0129	1.0005
4	.0983	.0765	.1857	.2798	.1857	.0765	.0713	.0270	1.0009
5	.0461	.0400	.0836	.1857	.2798	.2336	.0997	.0270	1.0005
6	.0759	.0837	.0960	.1836	.5608	0	0	0	1.0000
7	.0896	.0936	.1331	.2850	.3987	0	0	0	1.0000
8	.0978	.0987	.1544	.3246	.3246	0	0	0	1.0001

*See Figure A.32-1 for surface locations

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TABLE A.32-2

E1739A FORM FACTOR RESULTS FOR CASE 1

		TO SURFACE *						
		1	2	3	4	5	6	TOTAL
	1	0	.4829	.2410	.2001	.0790	.0085	1.0115
FORM	2	.3696	.2600	.2346	.0999	.0234	.0075	1.0000
FACTOR	3	.4408	.5608	0	0	0	.0030	1.0046
FROM	4	.6003	.3916	0	0	0	.0081	1.0000
SURFACE	5	.6581	.3089	0	0	0	.0330	1.0000
	6	.3197	.3675	.0624	.1018	.1486	0	1.0000

*See Figure A.32-1 for surface locations

Amend. 45
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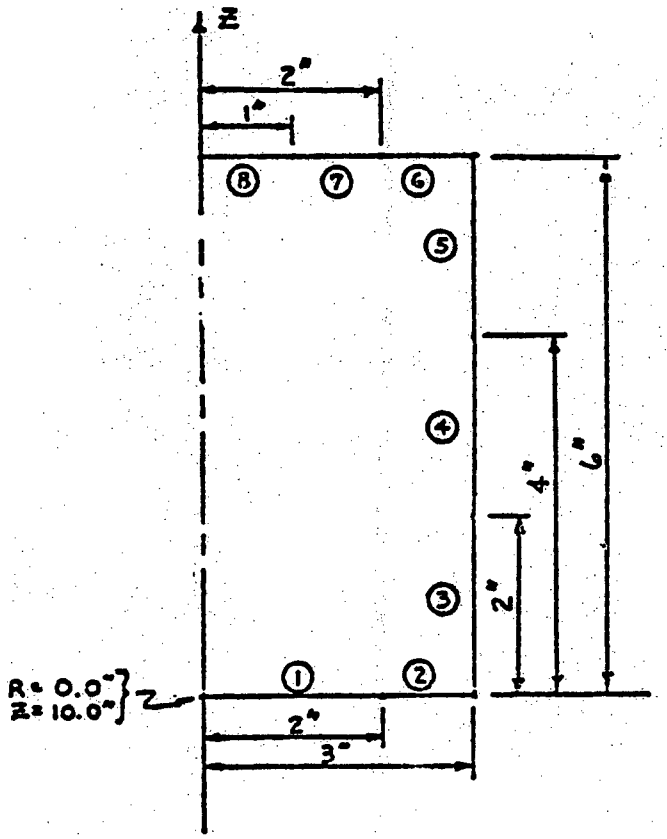

FIGURE A.32-1

AXISYMMETRIC CYLINDER

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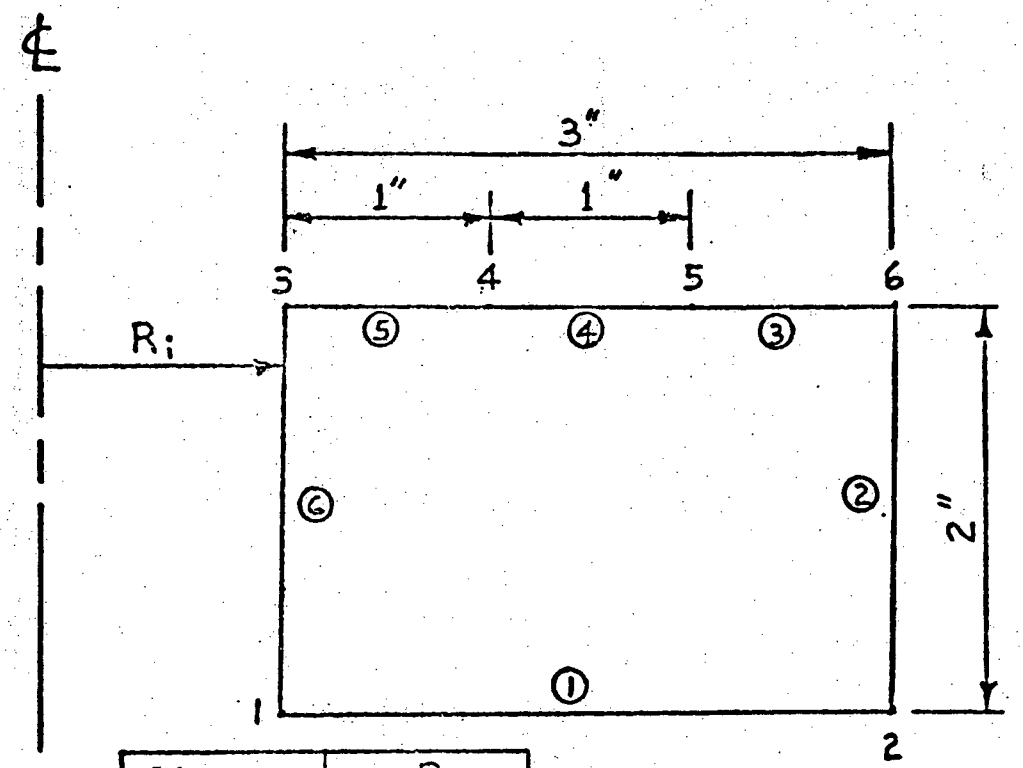

CASE	R_i
1	0.0625
2	2.0
3	5.0
4	10.0
5	50.0
6	100.0
7	300.0

○ = SURFACE NUMBER

FIGURE A.32-2

PROBLEM GEOMETRY

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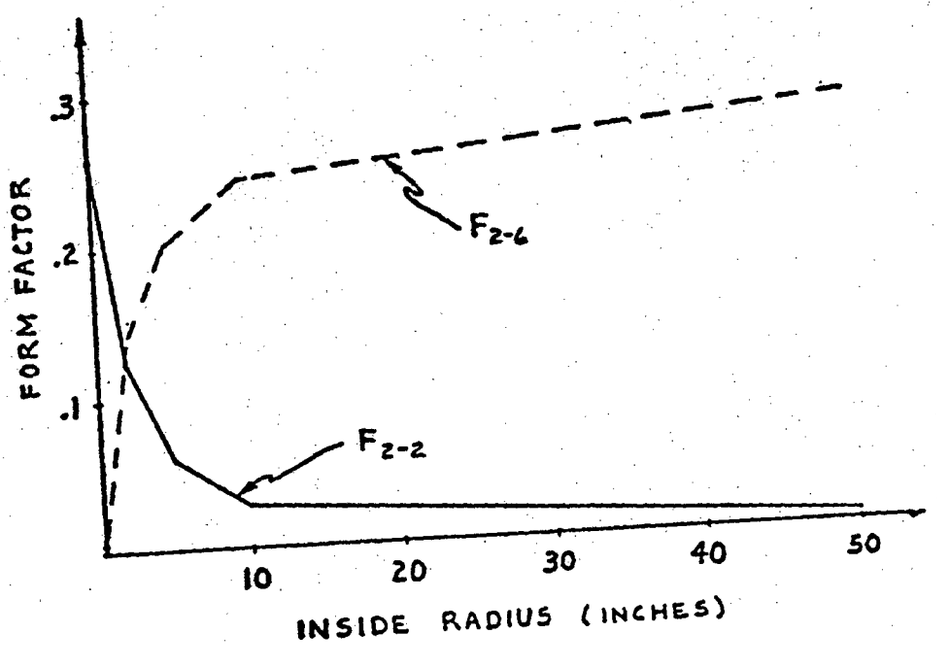

FIGURE A.32-3

FORM FACTOR RESULTS

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A.33 FATHØM-360 and FATHØM-360S

The FATHØM-360 codes calculate, at any axial position, the detailed coolant velocity and temperature profile around the rod circumference in LMFBR core assemblies, as well as the rod radial and circumferential temperature profile. The FATHØM-360 code analyzes an inboard rod, while the FATHØM-360S code analyzes a side rod. Fuel, blanket and control assemblies can be investigated; both wire wrapped and bare rods can be analyzed. The detailed coolant velocity distribution around the rod circumference is calculated along with the coolant, cladding and fuel (or absorber) temperature profile. Film and cladding circumferential hot spot factors due to subchannel geometry and wire wrap presence are calculated. Uniform and non-uniform heat generation (i.e., power skew across the pellet) in the fuel rod can be considered as well as both concentric and eccentric position of the fuel (or absorber) pellet within the cladding. The effect of inter-assembly heat transfer through the duct wall is accounted for in FATHØM-360S.

Availability

The FATHØM-360 codes are available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

Preceding versions of FATHØM have been successfully verified against other purely theoretical models, Figure A.33-1 (Fig. 4 from Reference 1) and theoretical models corroborated by experimental data, Figure A.33-2 (Fig. 5 from Reference 1), for the bare rods case. Also, excellent comparison was obtained against experimental data gathered in a 11:1 scale bare rod bundle air flow test at Westinghouse Research Laboratories, Figure A.33-3 (Fig. 6 from Reference 3). For wire wrapped rods, FATHØM-360 and 360S predictions have been compared against velocity profile experimentally determined at Westinghouse Research Laboratories in the 11:1 scale 1/6 sector mockup of CRBRP 217-pin fuel assembly. Typical results, which show a very good comparison, both qualitatively and quantitatively, of the complex velocity profile around the pin are reported in Figures A.33-4 and A.33-5. Further verification of FATHØM-360 against data from the radial blanket heat transfer test is planned, to verify temperature profile calculations in wire wrapped rods.

Application

FATHØM-360 and 360S are used in determining local hot spots in CRBRP core assemblies rods.

References

1. M. C. Chuang, R. E. Kothmann, M. J. Pechersky and R. A. Markley, "Cladding Circumferential Hot Spot Factors for Fuel and Blanket Rods", Nucl. Eng. and Design, 35, pp. 21-28 (1975).

2. M. C. Chuang, M. D. Carelli, C. W. Bach and J. S. Killimayer, "Three-Dimensional Thermal-Hydraulic Analysis of Wire Wrapped Rods in LMFBR Core Assemblies", Proceedings of ANS Topical Meeting on "Improved Methods for Analysis of Nuclear Systems", Tucson, Arizona, March 28-30, published in Nuc. Science and Eng., 64, No. 1, pp. 244-257 (September 1977).

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A.33, cont'd

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3. M. J. Perchersky, et. al., "11:1 Scale Rod Bundle Flow Tests - Parts 1 through 7" WARD-OX-3045-6, February 1974.
(Availability: USDOE Technical Information Center).

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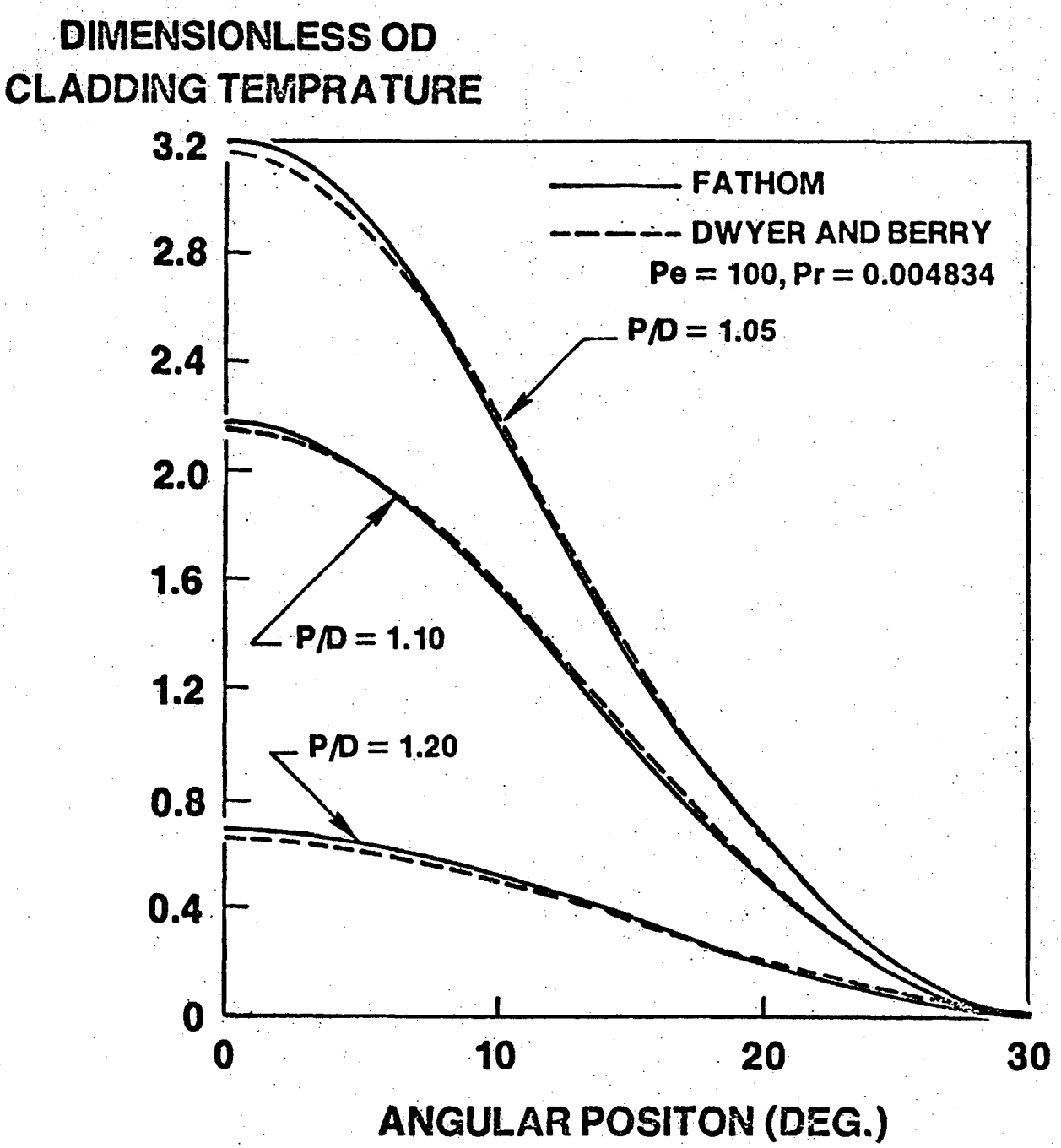
COMPARISON FATHOM/DWYER & BERRY FOR BARE RODS



S538-16

From Reference 1, FATHØM-360 code

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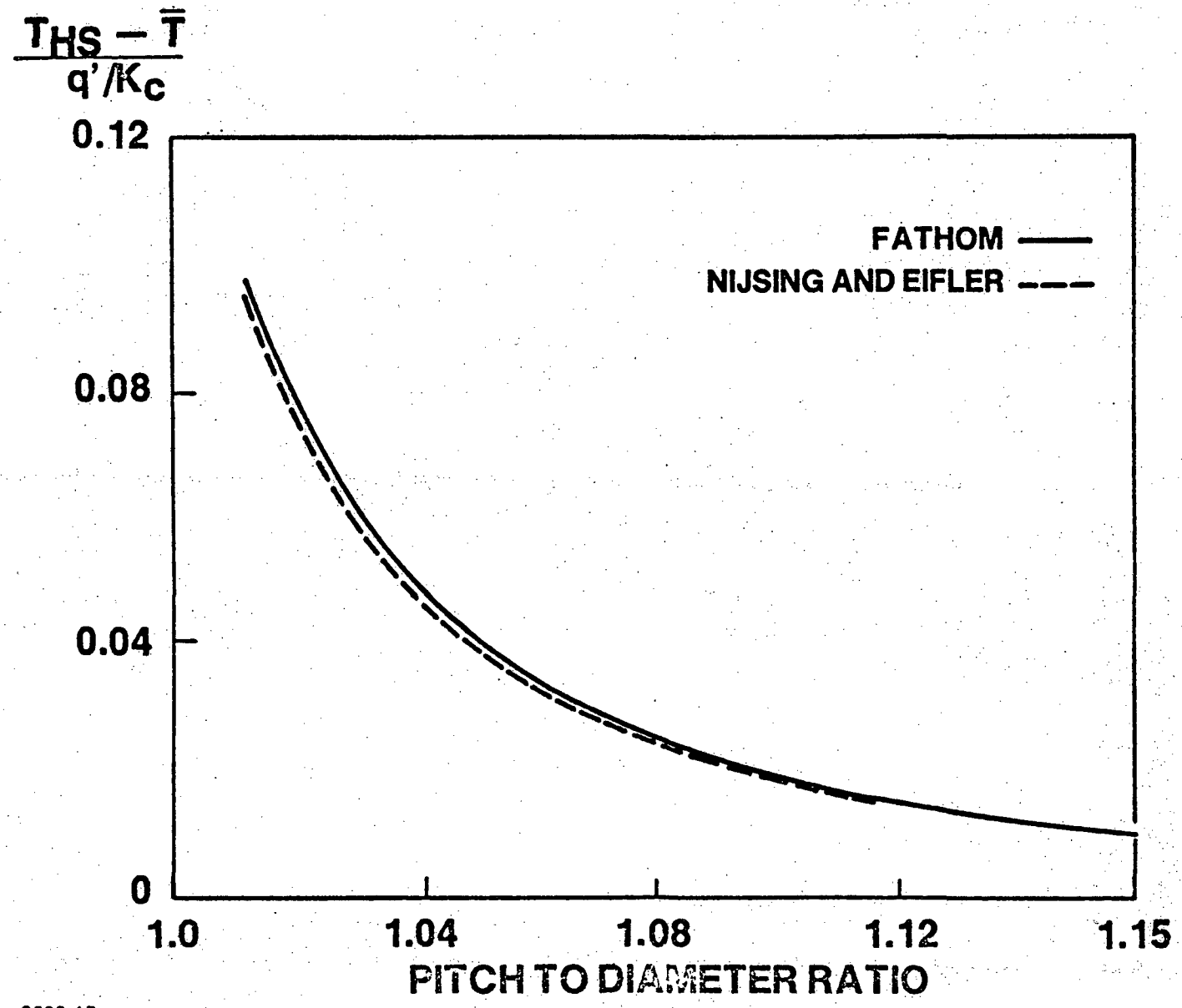
Figure A.33-1

COMPARISON FATHOM/NIJSING & EIFLER FOR BARE RODS



S538-17

From Reference 1, FATHOM-360 code

Figure A.33-2

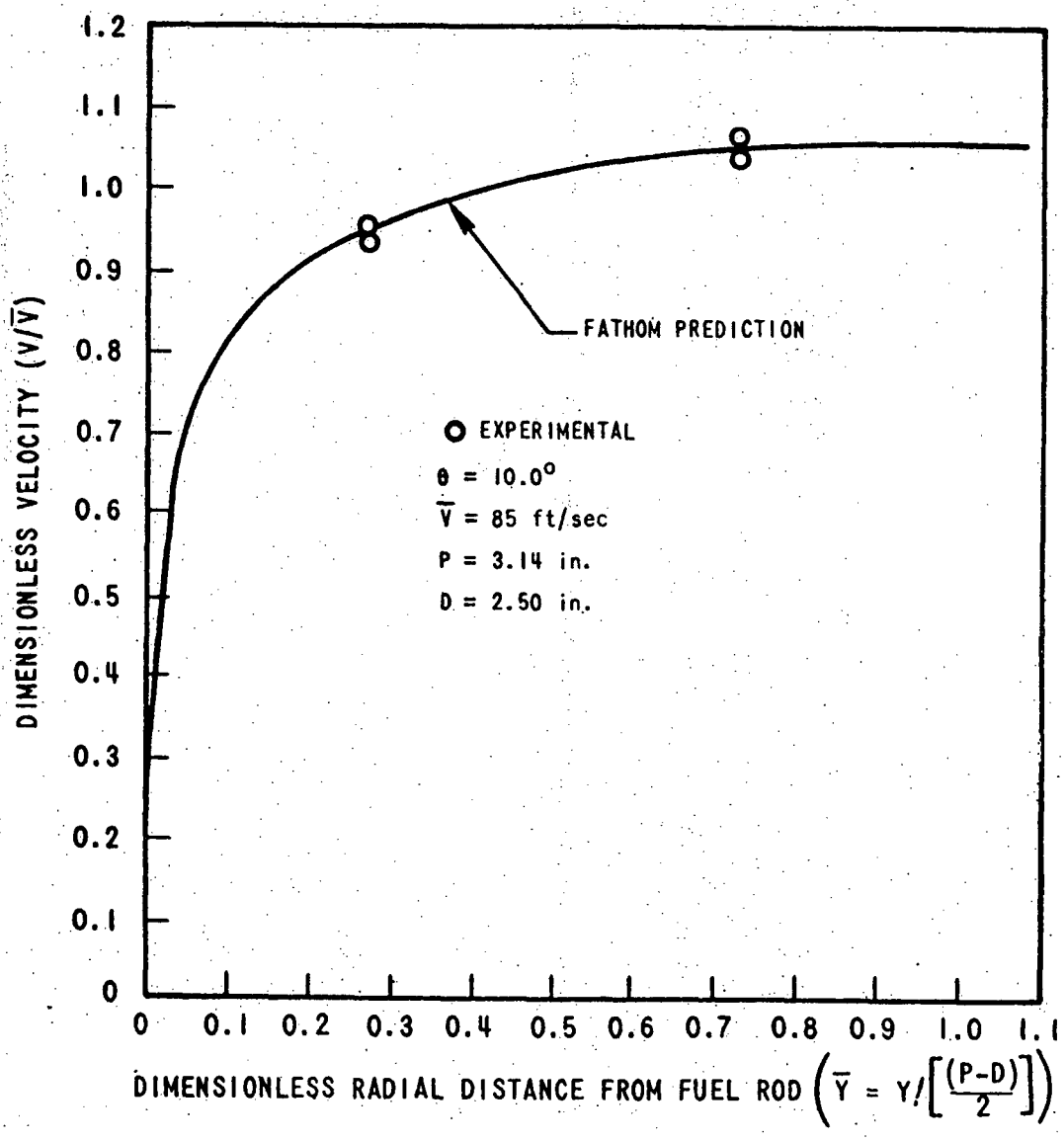


Figure A.33-3 Comparison of Experimental and Fathom Velocity Profiles in an Inboard Subchannel ($\theta = 10.0^{o}$)

6167-70

From Reference 3, FATHØM-360 Code

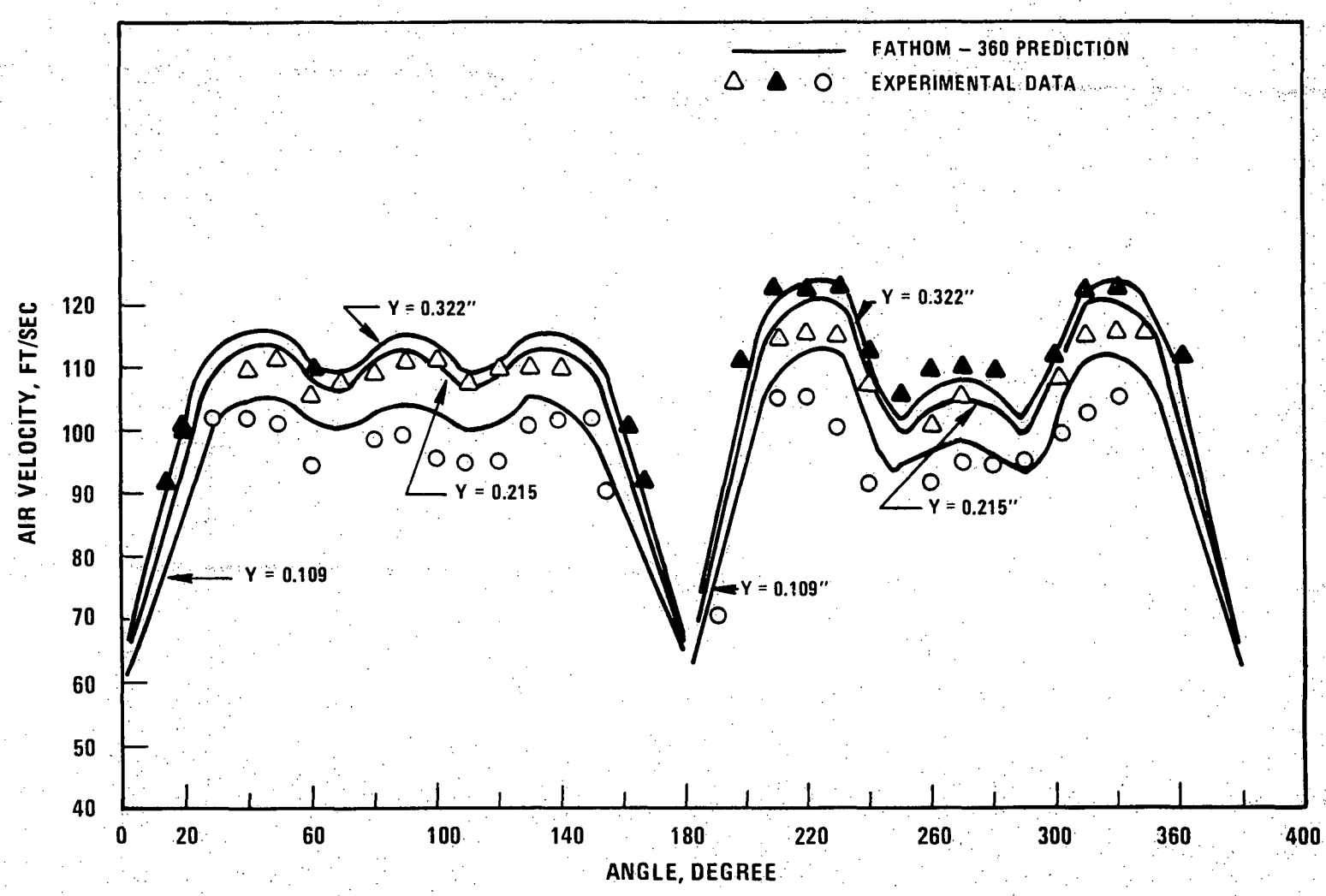
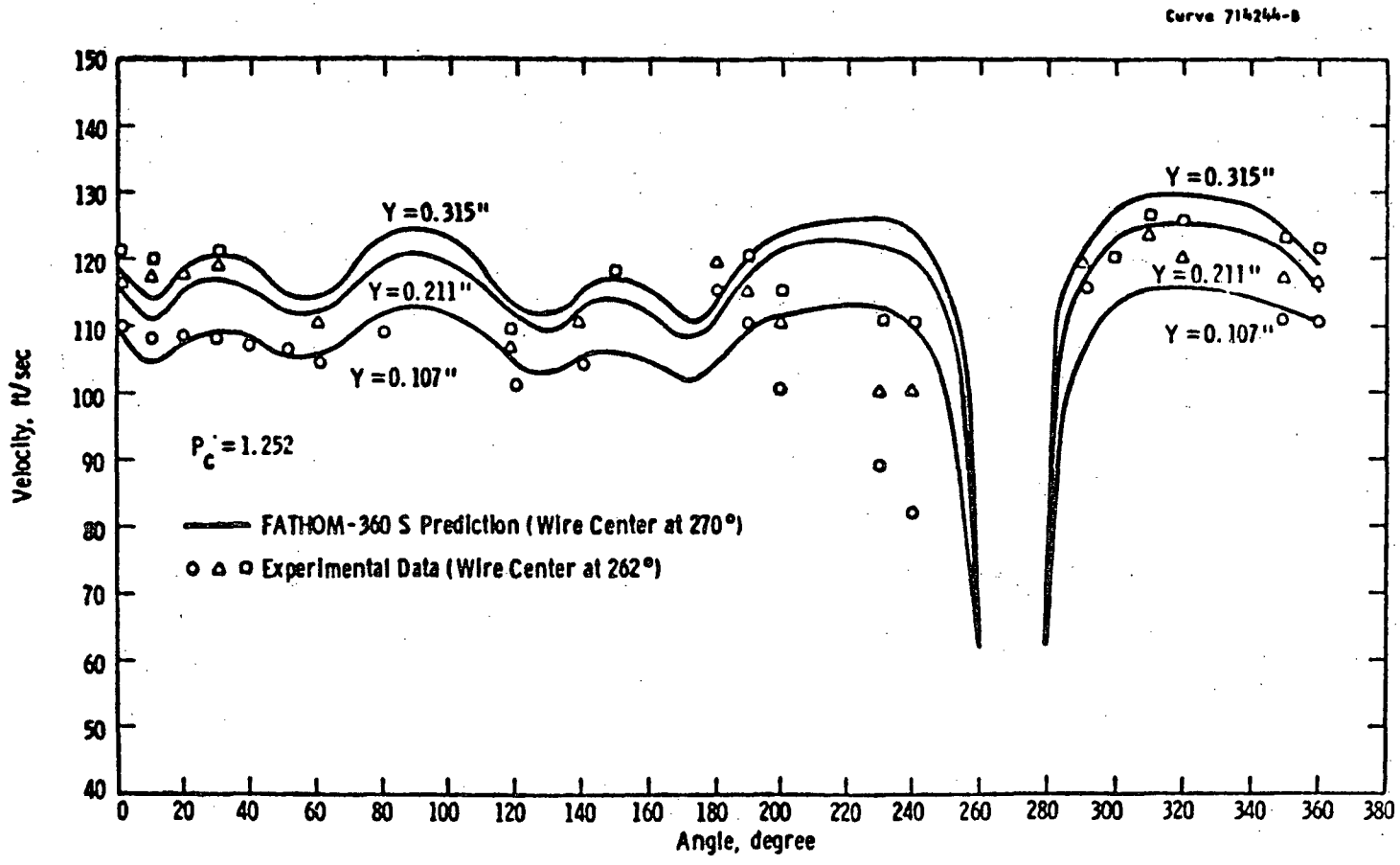


Figure A.33-4 Comparison of Theorizetical Predictions with Experimental Data in Composite Subchannels, 9 Inches from Downstream End of Pitote Tube Travel (Y = Radial Distance from the Clad Outer Surface)



Comparison of theoretical predictions with experimental data, 21 inches from downstream end of pitot tube travel
(Y = radial distance from the clad outer surface)

FIGURE A.33-5

A.34 FBRDSAP

FBRDSAP is a modified version of SAP IV. In time history analysis, FBRDSAP is able to save the displacement history data on a tape which can then be used to determine velocity history. This is the only difference from SAP IV.

Availability

FBRDSAP has been available on the CDC 7600 computer of Lawrence Berkeley Laboratory since February, 1975.

Verification

Verification of FBRDSAP is in accordance with criteria II.2.b of SRP 3.9.1. The computer program solutions to a series of benchmark problems (see Reference 1) with accepted results have been demonstrated to be identical to those obtained with SAP IV (A.76). A comparison of results is made in Figure A.34-1.

Application

FBRDSAP will be used for generating a structure's velocity history initiated by a seismic ground motion. The velocity so generated is used as input data to a hydrodynamic computer code, such as HYTRAN, for hydraulic transient analysis.

Reference

Bathe, K. J., Wilson, E. L. and Peterson, F. E. "SAP IV-A Structural Analysis Program for Static and Dynamic Response of Linear Systems," Report No. EERC 73-11, University of California, Berkeley, California, June 1973, Revised April 1974.

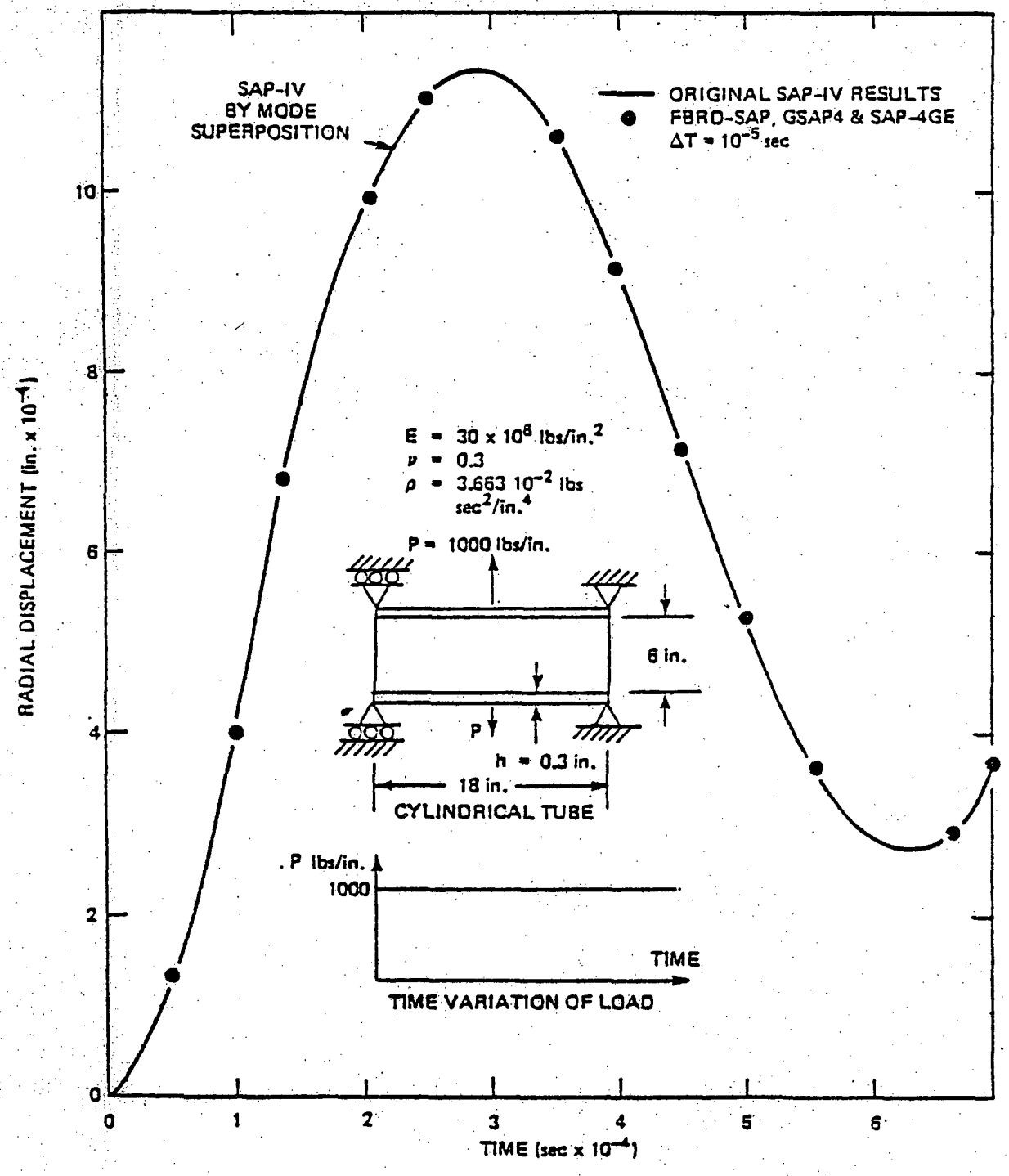


Figure A.34-1 *Comparison of FBRD-SAP and the SAP-IV Computer Code to a Benchmark Problem*

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A.35 FESAP (Babcock & Wilcox Proprietary)

FESAP, a proprietary Babcock & Wilcox computer program, is a 3-D finite element computer program capable of solving dynamic problems with large number of nodes.

Verification

The FESAP program has been verified satisfactorily with test cases reported in the literature and with dynamic problems within Babcock & Wilcox by several bench-mark problems, including two cases listed in the Reference. The FESAP program will also be verified by checking the FESAP program solution of a seismic test problem of a simple model, similar to the reactor vessel structural geometry, against that of ANSYS (A.3).

The verification is in accord with SRP Sections 3.9.1.II.2.b and c.

Application

FESAP will be used by Babcock & Wilcox Company in performing all dynamic seismic analyses of the total reactor vessel assembly.

Reference

"Pressure Vessel and Piping, 1972 Computer Program Verifications" American Society of Mechanical Engineers.

A.36 FLODISC

FLODISC (References 1 and 2) calculates the steady state flow distribution in a series of parallel flow channels connected by common plena at the inlet and outlet of each channel. The flow distribution is calculated assuming equal pressure differential across each channel. The effects of friction, contraction and expansion losses, elevation or buoyancy, and acceleration are included. Flow can either be upward or downward in a particular channel.

FLODISC, as currently used at ARD, includes minor modifications which allow the form and friction losses in the various assembly components to be modeled phenomenologically. As such, the accuracy of FLODISC predictions is chiefly limited by:

a) The accuracy of experimental data;

b) The accuracy of phenomenological correlations to experimental data;

Minor effects on the code accuracy are due to the convergence criterion used for the simultaneous solution of the system of phenomenological equations and the effect of inter-assembly heat transfer on assembly hydraulics.

Availability

The FLODISC code is available on the Westinghouse Pwer Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

The flow distribution predicted by FLODISC is a simple application of the mathematical theorem that the line integral of pressure between two points is independent of the path. The hydraulic characteristics input to FLODISC are the result of a simple definition of pressure drop in terms of observable properties, and hence, the definition needs no verification. What is required is to verify that the accuracy criteria listed above are met to a sufficient extent.

The accuracy of core assembly experimental data is discussed in Reference 3. Although data have not been gathered on every assembly component, as discussed in Reference 3, the accuracy and repeatability of data for other components can be expected to be similar. Hydraulic characterizations of additional assembly components at both design conditions and at low flow rates is continuing.

The accuracy of the phenomenological correlation to experimental data is generally excellent and lies well within the experimental data scatter. Typical fits are shown in Reference 4 and in Figures A.36-1 and 2. These figures also show the scatter typical of experimental hydraulic data.

The convergence criterion used for equation solution is trivial and can be made as small as necessary to eliminate its effect, while the effect of inter-assembly heat transfer on the calculation of reactor pressure drop versus flow rate can be calibrated out by comparison to detailed predictions.

Application

The FLØDISC code is used for calculating the reactor pressure drop as a function of reactor flow rate, using the phenomenological correlations for the various assemblies hydraulic characteristics. The result of this application is used for primary loop performance predictions, including natural circulation transients. The code also has been used to qualitatively predict inter-assembly flow redistribution, which was found to be significant at low flow rates due to buoyancy effects.

References

53 1. E. H. Novendstern, "FLØDISC: A Computer Code for Calculations of Flow Distributions in Parallel Channels", Westinghouse Advanced Reactors Division, Madison, PA, FRT-695, March 1972. (Availability: 53 USDOE Technical Information Center)

2. E. H. Novendstern, "Flow Distribution Within the FFTF at Low Flow 53 Rates", FRT-676, January 17, 1972. (Availability: USDOE Technical Information Center)

3. WARD-D-0050, Rev. 2, "CRBRP Assemblies Hot Channel Factors Preliminary Analysis", August 1977.

4. HEDL-TME-77-8, P. M. McConnel, "CRBR Fuel Assembly Inlet/Outlet Nozzle Flow Tests", February 1977.

5. HEDL-TC-824, W. L. Thorne, "Pressure Drop Measurements from Fuel 53 Assembly Vibration Test II", April 1977. (Availability: USDOE Technical Information Center)

6. HEDL-TI-76049, "Covered Pressure Drop Flow Text/Crossflow Mixing 53 Test", November 1976. (Availability: USDOE Technical Information Center)

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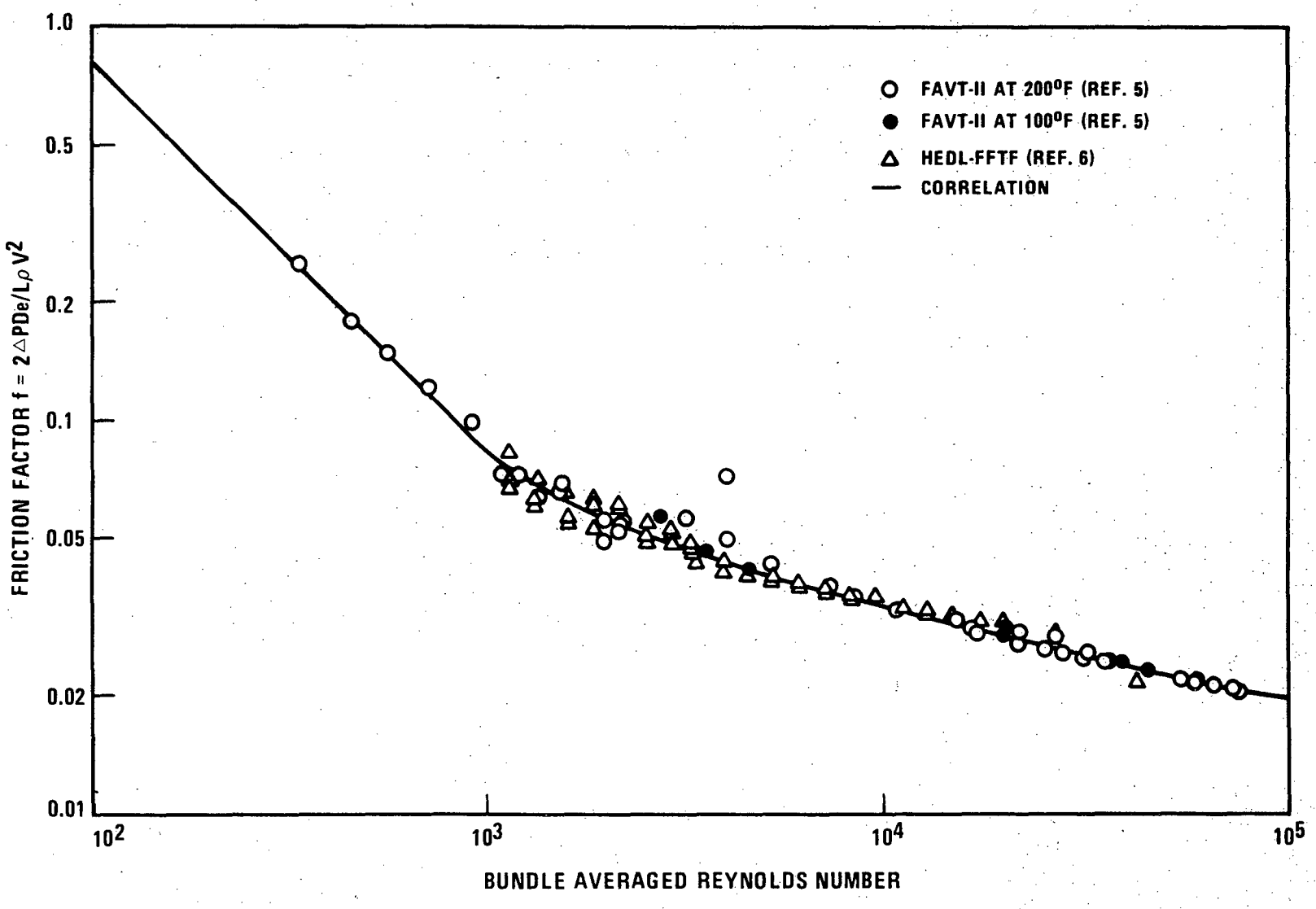

Figure A.36-1 Friction Factors in Wire Wrapped Rod Bundles Similar to CRBR Fuel Assemblies

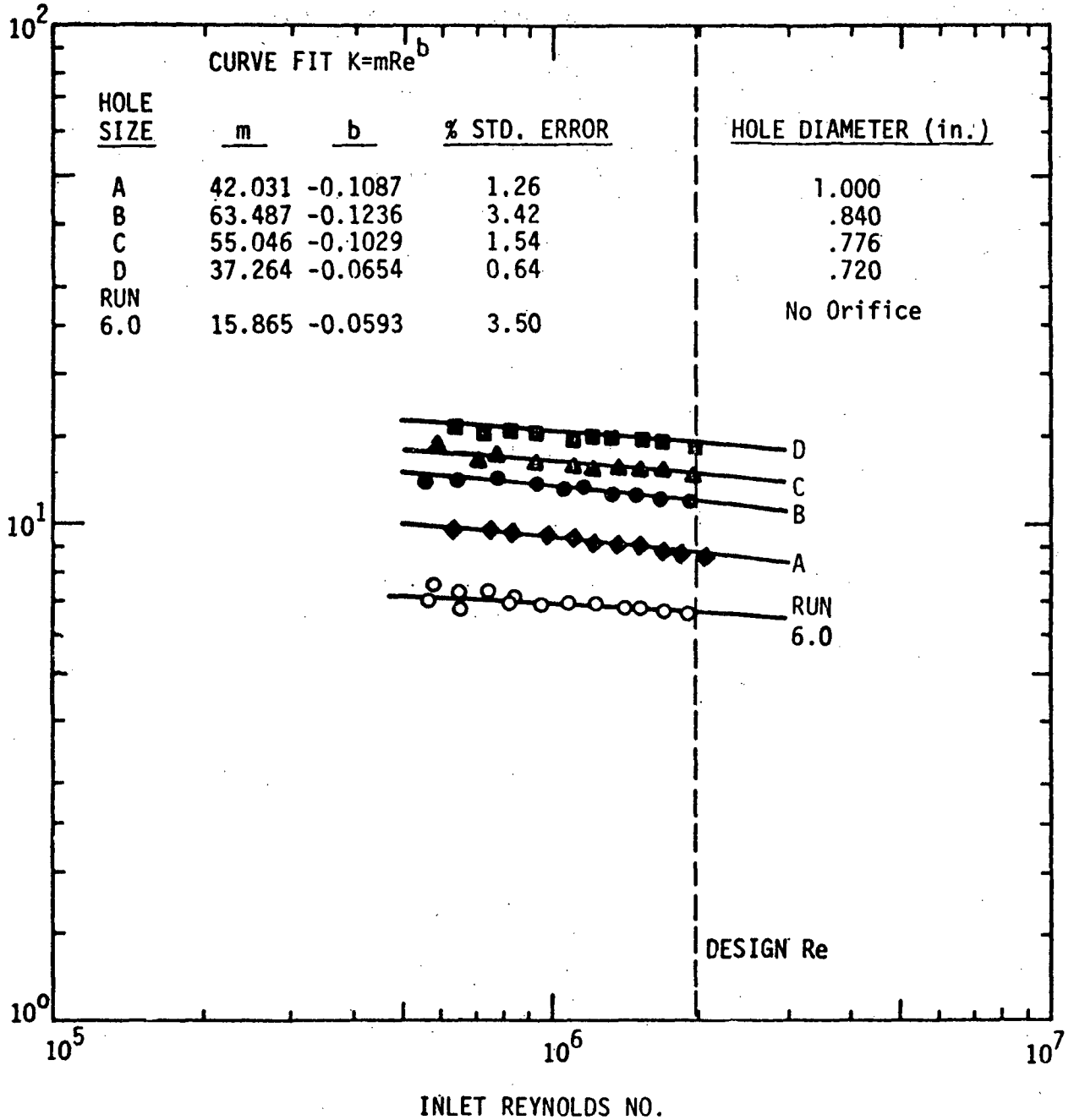

Figure A.36-2 **Fuel Assembly Inlet Nozzle Flow Resistance vs Reynolds Number; Single Plate Orifices and No Orifice Plate (from Reference 4).**

A.37 FORE-2M

FORE-2M, which is an improved version of FORE-II, is a coupled thermal-hydraulics point-kinetic digital computer code designed to calculate significant reactor care parameters under steady state conditions or as functions of time during transients. Variable inlet coolant flow rate and temperature are considered. The code calculates the reactor power, the individual reactivity feedbacks, and the temperature of coolant, cladding, fuel, structure, and additional material for up to seven axial positions. Various Plant Protection System trip functions can be simulated, and the control rod shutdown worth prescribed as a function of time from the trip signal. By specifying appropriate hot channel/spot factors, the transient behavior of an average, peak and hot fuel rod can be analyzed. The heat of fusion accompanying fuel melting and the spatial/time variation of the fuel-cladding gap coefficient (due to changes in gap size) are considered. The feedback reactivity includes contributions due to the Doppler effect, coolant density changes and dimensional changes (including bowing and radial expansion). FORE-2M is valid only while the core retains its initial geometry.

The original FORE-II computer model (Reference 1) was renamed FORE-2M following the incorporation of several major changes which were made to the program (Reference 2). Since then, additional modifications have been made to the code. These include updated modeling of the gap conductance heat transfer, changes affecting material properties, modifications in transient coolant flow characteristics, simulation of inter- and intra-assembly flow and heat redistribution, reactivity feedback and decay heat modifications, model changes to allow for alternate fuel rod characteristics and program improvements to provide user flexibility. These changes are described in Reference 3 which also provides the required input variables associated with these modifications.

Availability

The FORE-2M code described in Reference 2 and 3, is available on the Westinghouse Power Systems CDC-7600 computers and CRAY-1 computers located at the Monroeville Nuclear Center.

Verification

In addition to comparisons with closed form analytical solutions, FORE-2M transient results have been compared to core data from EBR-II natural circulation experiments (Ref. 4) as well as natural circulation experiments performed in FTR (Ref. 5). The original FORE-II code has been used extensively over the last 17 years in the nuclear industry.

Application

The FORE-2M code is used to calculate the nuclear kinetic response of the core as well as the average, peak and hot rod behavior at steady state conditions, or as a function of time during transients.

References

1. J. N. Fox, B. E. Lawler, H. R. Butz, "FORE-II; A Computational Program for the Analysis of Steady State and Transient Reactor Performance," GEAP-5273, September, 1965.

2. J. V. Miller, R. D. Coffield, "FORE-2M: A Modified Version of the FORE-II Computer Program for the Analysis of LMFBR Transients," WARD-D-0142, May, 1976.

3. J. V. Miller, R. D. Coffield, K. D. Daschke, et. al., "Supplementary Manual for the FORE-2M Computer Program," CRBRP-ARD-0257, September 1982. Amend. 45, July 1978.

4. Y. S. Tang, R. D. Coffield and F. C. Luffy, "Verification of FORE-2M Computer Code Part II. Comparison with EBR-II Natural Circulation Experiments," CRBRP-ARD-0315, October 1982.

5. A. C. Cheung, R. D. Coffield, K. D. Daschke, and Y. S. Tang, "Verification of the CRBRP Natural Circulation Core Analysis Methodology with Data from FFTF Natural Circulation Tests," CRBRP-ARD-0310, June 1982.

A.38 FRST

The FRST code calculates cladding strains due to steady state and transient conditions. This code considers the effects of internal gas pressure, and the localized effects of temperature and fluence. The cladding wall thickness reduction due to fretting wear, sodium corrosion, and fuel-cladding interaction is also considered. The statistical uncertainty in environment and material properties is considered by using the conservative upper or lower design limits on material properties and the conservative environments at design conditions. The calculated total ductility limited hoop strain includes the sum of the plastic strain, primary thermal creep strain, and secondary creep strain. The elastic strain. volumetric swelling strain and irradition-induced creep strain components are calculated to determine the total axial, hoop and radial deformations. It is assumed that early in life the thermal stresses are relaxed by plasticity and thermal creep. This thermal strain component is, therefore, included in the total ductility limited strain. The Sodorburg relation is used to determine the magnitude of the thermal creep strain rate in a given direction and the equivilent stress state is determined by the Von Mises criterion. The end-of-life creep and swelling strains are obtained by numerical integration over the lifetime considering all time-varying dimensions, temperatures, stresses, fluences and creep rates.

Availability

The FRST code is non-proprietary, and is available from the Fuel and Removable Assembly Design Group at Westinghouse Advanced Reactors Division. It is written for the Sigma-5 computer at Westinghouse ARD.

Verification

The thermal Creep, irradiation creep and swelling, and wastage rates calculated by FRST have been compared to hand calculations. Results of these comparisons for two checkout problems with environments typical of CRBRP core conditions are given in Table A.38-1. Preliminary verification results and plans for further studies to verify the strain limit procedure to predict fuel rod design lifetime are discussed in PSAR Section 4.2.1.3.

Application

The FRST code is used to calculate the fuel and radial blanket rod cladding deformation components due to the in-core thermal and nuclear environments as a function of time. This code is also used to calculate the time at which the cladding reaches the appropriate ductility limited strain limit, thereby giving a measure of cladding service lifetime.

TABLE A.38-1

COMPARISON OF FRST COMPUTER CODE AND HAND CALCULATION RESULTS

Check Problem Conditions	Case 1	Case 2
Cladding Temperature (°F)	1200	1100
Plenum Pressure (psi)	400	500
Maximum Burnup (MWd/MT)	80,000	80,000
Fast Fluence (n/cm^2)	8×10^{22}	6×10^{22}
Sodium Velocity (ft/sec)	20	20
C_2 Content (ppm)	5	5
Cladding Mean Diameter (in.)	.215	.215
Cladding Thickness (in)	.015	.015
Lifetime (days)	300	300

	Case 1		Case 2	
Problem Results	FRST	Hand Calc.	FRST	Hand Calc.
Stress-Free Irradiation Swelling (in./in.)	4.87×10^{-3}	4.86×10^{-3}	9.08×10^{-3}	9.09×10^{-3}
Stress Enhanced Swelling (in./in.)	1.38×10^{-4}	1.33×10^{-4}	3.15×10^{-4}	3.11×10^{-4}
Secondary Irradiation Creep (in./in.)	2.51×10^{-3}	2.46×10^{-3}	2.93×10^{-3}	2.99×10^{-3}
Secondary Thermal Creep (in./in.)	1.343×10^{-6}	1.406×10^{-6}	3.32×10^{-10}	2.94×10^{-10}
Cladding Sodium Corrosion (in.)	.000345	.000301	.000155	.000169
Fuel-Cladding Corrosion (in.)	.00146	.00145	.001015	.001013

A.39 FULMIX

This code is a thermal-hydraulic interchange mixing code for hexagonally arranged pin bundles with sodium coolant. The code performs the steady-state calculation of average temperatures for each fuel rod and coolant flow cell in the pin bundle at an arbitrary number of planes. Flow splits between cells are calculated from input dimensions assuming no radial pressure gradients. The magnitude of turbulent mixing between adjacent cells can be varied from zero (only molecular conduction) to values representative of wire-wrapped bundles.

Availability

FULMIX is written in the FORTRAN language and is operational on the Honeywell 6000 computer. The code is non-proprietary.

Verification

FULMIX has been compared to the COBRA computer code and is in good agreement when there is no large flow redistribution due to blockages. The documentation of the comparison is in the reference. the COBRA code has been compared to experiments and other codes performing similar calculations.

Application

FULMIX can be used to determine the coolant temperature distribution throughout a pin bundle. It is used to identify the pin having the peak cladding temperature and to evaluate the effects of geometry changes (such as wire wrap pitch, edge pin spacing, pin pitch) on the coolant and cladding temperatures.

Reference

Magee, P.M., "FULMIX - Turbulent Interchange Mixing Code for Fuel Bundle Thermal-Hydraulic Analysis," GE FBRD Core Development Memo 150-13, June 1971.

A.40 FURFAN

FURFAN is a FORTAN code which applies the Cumulative Mechanical Damage Function (CDF) to fuel-pin performance analysis. The code computes the fuel-pin failure time based on the steady state operating history combined with a varied number of transient events each of which may be taken to occur at a specified number of times throughout life. The general procedure treats the effects of stress, temperature, prior mechanical history, corrosion, interstitial loss and fisson product attack.

The models describing the mechanical properties and behavorial characteristics of the clad are in subroutines and are mutually compatible and consistent. All models are based on experimental data which have been fit by linear regression techniques. Therefore, in additon to representing each property by an efficient mathematical formulation, each model is characterized by a set of statistical parameters which are used to establish analytical uncertainties.

Availability

The FURFAN code is non-proprietary, and is available on the Westinghouse Power System CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

FURFAN and the CDF procedure have been verified experimentally using carefully controlled laboratory tests with irradiated and un-irradiated material (See Reference Below). In all cases the CDF procedure was capable of predicting, within the limits of analytical uncertainty, the observed experimental failure. The range of environments over which the FURFAN is valid is being expanded and updated as new data becomes available. Verification against integral fuel rod test data is planned, and is discussed in PSAR Section 4.2.1.3.

Application

The FURFAN code is being used to calculate the lifetime of fuel and radial blanket rods in the CRBRP.

Reference

CRBRP-ARD-0115, "The Development and Application of a Cumalative Mechanical Damage Function for Fuel-Pin Failure Analysis in LMFBR System", D. C. Jacobs, May, 1976. (Availability: USERDA Technical Information Center).

A.41 GAMLEG-W

GAMLEG-W, a revised version of the GAMLEG code, provides multigroup, photon transport cross sections for use in multigroup, discrete ordinate transport or Monte Carlo transport codes. The GAMLEG-W code is designed to provide photon transport cross sections in a maximum of 100 groups with scatter transfer cross sections represented as a Legendre expansion ($P\ell$) of arbitrary order. The code performs a numerical integration with a user specified weighting function to obtain multigroup, absorption and scatter-transfer cross sections. Absorption cross sections are obtained from pointwise, photoelectric and pair-production data on punched data cards or magnetic tape, and Compton absorption from the Klein-Nishina equation for the inelastic scattering of a photon with a free electron. Scatter-transfer cross sections are obtained from the differential form of the Klein-Nishina equation for the inelastic scattering of a photon with free electron. The code, can, at option, provide photon transport cross section data for use in energy flux or particle flux calculations. Production of a pair of 0.511 Mev photons due to pair production annihilation can be included in the energy scatter-transfer of particle scatter-transfer cross section data. Output data from the GAMLEG-W code are compatible with ANISN (A.2) or DOT (A.22) computer codes.

Availability

The GAMLEG-W code is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center. The current version was released from Westinghouse Astronuclear Laboratory in August, 1970, and has been updated at ARD to satisfy CRBRP shield design analysis requirements.

Verification

The GAMLEG series of codes have been compared to: 1) analytic solutions of the photon cross sections, 2) hand calculations, and 3) comparisons to experiments. A summary comparison of the GAMLEG results versus the exact solution is shown in Figure A.41-1. The ANISN-W and DOTIIIW codes, using GAMLEG-W photon cross section data as input, have been used to predict gamma ray experimental data (spectra and heating).

Application

GAMLEG-W is an integral part of the CRBRP shielding design analysis method. Gamma ray cross sections produced by GAMLEG-W are used or processed for use in ANISN-W or DOTIIIW analysis of gamma ray flux environments in the CRBRP reactor system. In addition, gamma ray heating, biological shield attenuation, and gamma ray dose rates are predicted by use of GAMLEG-W data. Typical uses of GAMLEG-W data are prediction of gamma heating in CRBRP reactor system components, biological shields, and gamma dose rates at surface of biological shields, such as the CRBRP closure head assembly.

References

1. R. G. Soltes, R. K. Disney, and S. L. Zeigler, "Nuclear Rocket Shielding Methods, Modification, Updating, and Input Data Preparation. Volume 3. Cross Section Generation and Data Processing Techniques. Final Progress Report," WANL-PR(LL)-34, Vol. 3 (NASA-CR-102966), August 1970.

2. R. G. Jaeger, "Shielding Fundamentals and Methods," Engineering Compendium on Radiation Sheilding, Vol. I, pg. 190, Springer-Verlag, NY, 1968.

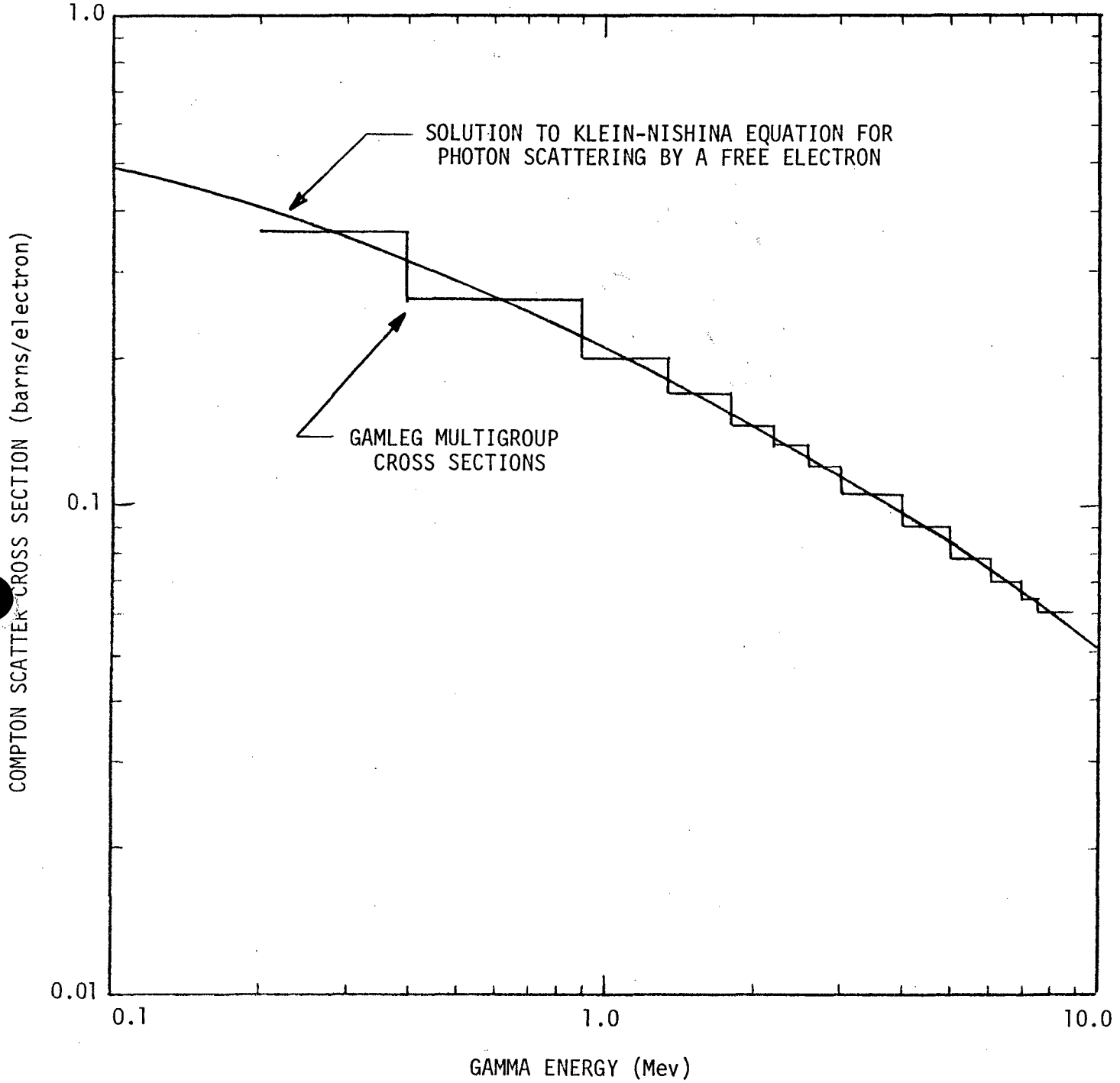

Figure A.41-1 COMPARISON OF MULTIGROUP PHOTON SCATTER CROSS SECTIONS WITH ANALYTIC COMPTON SCATTER CROSS SECTION (REFERENCE 2)

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(Next page is A-123)

A.43 GASP

The GASP program uses the finite element method to determine the stresses and displacements of plane or axisymmetric structures of arbitrary geometry.

The structures may be arbitrary geometry and have linear or non-linear material properties. The loadings may be thermal, mechanical, accelerational or a combination of these. The actual loads on the structure are simulated by statically equivalent loads acting at the appropriate nodes.

The Chicago Bridge and Iron Company's version of GASP has been expanded from the original version to include two special features beyond the basic solution described above. These features are the calculation of stresses on the surface of the model and the calculation of equivalent shell type (membrane and membrane plus linear bending) stresses.

Availability

This program is available on the IBM 370, Model 165 Computer of the Chicago Bridge and Iron Company.

Verification

Verification of this program is achieved through the use of the following problems:

1. Plane Strain Solution for a Cylinder with a Radial Temperature Gradient

   The problem considered is an 84 inch O.D. by 76 inch I.D. cylinder 60 inches long with an inside temperature of 100°F and an outside temperature of 500°F.

   The results are shown in Table A.43-1, radial displacements, Table A.43-2, surface stresses and element stresses; Table A.43-3, shell force resultants. The agreements between the elasticity solution, taken from Timoschenko and Goodier, "Theory of Elasticity," Third Edition, McGraw Hill Book Co., page 372, and the GASP Solution are good.

2. Circular Flat Plate with Edge Bending

   The problem considered is a circular flat plate with a diameter of 120 inches and a thickness of two inches. A uniform moment of 1000 in-lb/radian is applied at the edge of the plate.

   The results of the GASP program are compared with elasticity results using Roark "Formulas for Stress & Strain" Fourth edition, McGraw Hill Book Co., page 219, and shown in tables A.43-4 and A.43-5.

3. Cantilever Cylinders with Thermal Gradient

The problem consists of a 42 inch O.D. by 38 inch I.D. cylinder 60 inches long. The inside surface of the cylinder is at $0^{o}F$ while the outside surface is at $400^{o}F$. The cylinder is fixed at one end to prevent deflection and is free at the other end.

The GASP results are compared to results obtained from kalnins and are tabulated in Tables A.43-6 and A.43-7.

4. Cantilever Cylinder with Axial Acceleration

The problem consists of a 42 inch O.D. by 38 inch I.D. cylinder 60 inches long. The cylinder is loaded axially with an acceleration of 100g. The cylinder is mounted on rollers at one end so that it can deflect radially but not axially. The other end is free.

The GASP's results are compared to results obtained from kalnins. Deflections aretabulated in Table A.43-8. Stresses are tabulated in Table A.43-9.

5. Partial Spherewith External Pressure

The problem considered is an 84 inch O.D. by 76 inch I.D. sphere subjected to an external pressure of 10,000 psi.

The elasticity solution from Roark "Formulas for Stress & Strain" Fourth Edition, McGraw Hill Book Co., page 308, are compared to the GASP results from points as close as possible to the boundary. Deflections are listed in Table A.43-10. Stresses are listed in Table A.43-11. The shell force resultants are listed in Table A.43-12 and are in close agreement.

6. Feedwater Nozzle

The problem considered is a feed water nozzle mounted in a spherical section of a reactor vessel. The feed water nozzle consists of a main nozzle forging, safe end, safe end extension. The entire area considered has a change in temperature of $100^{o}F$.

The results of the GASP program are compared with theoretical problem taken from Levinson, "Mechanics of Materials", Prentice Hall Inc., page 45, and the deflections are listed in Table A.43-13.

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Reference

Wilson, E. L.; "A Digital Computer Program for the Finite Element Analysis of Solids with Non-Linear Material Properties" Aerojet General Corporation, Sacramento, California. Technical Memorandum No. 23, July 1965.

Table A.43-1

Radial Deflections

R	ELASTICITY SOLUTION	NODE	GASP	% ERROR	NODE	GASP	% ERROR
38	.1124438	5	.1123207	0.1	.50	.1123164	0.1
39	.1134529	4	.1131715	0.2	49	.1131670	0.2
40	.1157995	3	.1154493	0.3	48	.1154445	0.3
41	.1197591	2	.1191175	0.5	47	.1191121	0.5
42	.1242800	1	.1241431	0.1	46	.1241369	0.1

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Table A.43-2

Stresses - Elasticity Solution

R	Z	σ_R	σ_Z	σ_T	σ_{RZ}
16.06	38.77	-54013	-9267	-131485	22373
15.87	38.30	-41018	-7037	-114936	16990
15.48	37.38	-13918	-1162	- 83285	6378
15.10	36.46	13186	3536	- 51484	4285
14.72	35.54	41568	7762	- 18875	16903
14.53	35.07	55514	9525	- 3165	22995

Stresses - GASP Solution

R	Z	σ_R	% ERROR	σ_Z	% ERROR	σ_T	% ERROR	σ_{RZ}	% ERROR
16.06	38.77	-54016	.006	-9266	.01	-131480	.004	22372	.004
15.87	38.30	-40722	0.7	-6528	7.8	-115420	0.4	17093	0.6
15.48	37.88	-13845	0.5	-1235	6.3	- 83274	0.01	6303	1.2
15.10	36.46	13571	2.9	3519	0.5	- 51123	0.7	5026	4.2
14.72	35.54	41587	0.5	7676	1.1	- 18971	0.5	16949	0.3
14.53	35.07	55740	0.4	9552	0.3	- 2912	8.7	23075	0.3

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Table A.43-3

SHELL FORCE AND MOMENT RESULTANTS

	N_θ	$N_{\theta z}$	Q_θ	M_θ	$M_{\theta z}$	N_z	$N_{z\theta}$	M_z	$M_{z\theta}$
ELASTICITY SOLUTION	0	0	0	-171092	0	-272820	0	-159992	0
GASP	0	0	-1	-168817	0	-272540	0	-160348	0
% ERROR	-	-	-	1.3	-	0.1	-	-.2	-

TOTAL FORCE	NORMAL TO PLANE
ELASTICITY	-8570893
GASP	-8562097
% ERROR	0.1

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Table A.43-4

Deflection at Neutral Axis of Plate Thickness

RADIUS(in)	10	20	30	40	50	60
DEFLECTION(in) ELASTICITY	$2.91667x^{-5}$	$1.16667x10^{-4}$	$2.625x10^{-4}$	$4.66667x10^{-4}$	$7.29167x10^{-4}$	$1.05x10^{-3}$
DEFLECTION(in) GASP	$3.11033x10^{-5}$	$1.19882x10^{-4}$	$2.64353x10^{-4}$	$4.63431x10^{-4}$	$7.16367x10^{-4}$	$1.02259x10^{-3}$
% DIFFERENCE	6.6	2.7	0.7	0.7	1.8	2.7

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Table A.43-5

STRESSES

Elasticity Solution

Distance y	1.0	0.75	0.25
Radial & Tangential Stress	25.00	18.75	6.25

GASP Solution

RADIUS	20.5			30.5			40.5			50.5		
	y	STRESS	% DIFF	y	STRESS	% DIFF	y	STRESS	% DIFF	y	STRESS	% DIFF
RADIAL STRESS	1.00	24.093	3.8	1.00	23.644	5.7	1.00	23.328	7.2	1.00	23.086	8.3
	0.75	18.070	3.8	0.75	17.733	5.7	0.75	17.496	7.2	0.75	17.314	8.3
	0.25	6.0235	3.8	0.25	5.9111	5.7	0.25	5.8322	7.2	0.25	5.7715	8.3
TANGENTIAL STRESS	1.00	24.732	1.1	1.00	24.270	3.0	1.00	23.946	4.4	1.00	23.697	5.5
	0.75	18.548	1.1	0.75	18.203	3.0	0.75	17.960	4.4	0.75	17.773	5.5
	0.25	6.1827	1.1	0.25	6.0675	3.0	0.25	5.9865	4.4	0.25	5.9243	5.5

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Table A.43-6

COMPARISON OF DEFLECTION RESULTS

LOCATION		R DEFLECTION			Z DEFLECTION		
R	Z	GASP	KALNINS	% DIFF	GASP	KALNINS	% DIFF
20	0	0	0	0	0	0	0
20	10	.02928149	.02823	3.7	.01741545	.01734	0.4
20	20	.03164376	.03071	3.0	.03206377	.03224	0.5
20	30	.03094958	.03001	3.1	.04679741	.04721	0.9
20	40	.03091490	.03007	2.8	.06154796	.06219	0.1
20	50	.02717455	.02586	5.1	.07653177	.07742	1.2
20	60	.05115704	.05360	4.8	.09121978	.09221	1.1

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Table A.43-7

COMPARISON OF STRESS RESULTS

LOCATION		Z - STRESS			T - STRESS		
R	Z	GASP	KALNINS	% DIFF	GASP	KALNINS	% DIFF
21.00	20.50	63067	63845	1.2	64127	65095	1.5
20.75	20.50	47053	47882	1.8	48654	49057	0.8
20.25	20.50	15022	15961	6.3	17386	16983	2.4
19.75	20.50	-17025	-15964	6.6	-14644	-15095	3.1
19.25	20.50	-49054	-47885	2.4	-47510	-47173	0.7
21.00	30.5	63036	63995	1.5	63257	64225	1.5
20.75	30.5	47025	48018	2.1	47773	47995	0.5
20.25	30.5	15007	15996	6.6	16480	16076	2.5
19.75	30.5	-17010	-15999	6.3	-15574	-16024	2.9
19.25	30.5	-49024	-48000	2.1	-48468	-48122	0.7

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Table A.43-8

COMPARISON OF DEFLECTION RESULTS

LOCATION		R DEFLECTION			Z DEFLECTION		
R	Z	GASP	KALNINS	% DIFF	GASP	KALNINS	% DIFF
20	0	.00033291	.0003256	2.2	.00014783	0	-
20	10	.00028427	.0002854	0.4	.00061830	.0005186	19.2
20	20	.00022640	.0002264	0.0	.00104287	.0009434	10.5
20	30	.000169777	.00016976	0.0	.00137303	.0012734	7.8
20	40	.000113199	.00011320	0.0	.00160886	.0015092	6.6
20	50	.000056585	.00005660	0.0	.00175036	.0016508	6.0
20	60	.000000276	.000000783	1.0	.00179754	.0016979	5.9

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Table A.43-9

COMPARISON OF STRESS RESULTS

LOCATION		Z - STRESS			T - STRESS		
R	Z	GASP	KALNINS	DIFF %	GASP	KALNINS	DIFF %
20.75	10.50	1400.0	1399.8	0.0	1.9261	1.8638	3.3
20.25	10.50	1400.6	1400.4	0.0	1.8253	1.7583	3.8
19.75	10.50	1401.2	1401.2	0.0	1.7169	1.6530	3.9
19.25	10.50	1401.7	1401.8	0.0	1.6096	1.5480	4.0
20.75	40.5	551.84	551.9	0.0	NEGLIGIBLE		
20.25	40.5	551.85	551.9	0.0			
19.75	40.5	551.85	551.9	0.0			
19.25	40.5	551.86	551.9	0.0			

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Table A.43-10

RADIAL DEFLECTIONS

R	ELASTICITY SOLUTION	GASP NODE	GASP SOLUTION	% DIFF
38	.04757640	1	.04799253	0.9
39	.04850196	2	.04877145	0.6
40	.04942752	3	.04964266	0.4
41	.05035308	4	.05061700	0.5
42	.05127864	5	.05170186	0.8

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Table A.43-11

Stresses - Elasticity Solution

R	Z	σ_R	σ_Z	σ_T	σ_{RZ}
1.84	41.92	-52752	-10081	-52833	1867
1.82	41.42	-53271	-9039	-53355	1935
1.77	40.42	-54388	-6799	-54479	2082
1.73	39.42	-55621	-4326	-55719	2244
1.69	38.43	-56986	-1589	-57092	2423
1.66	37.93	-57723	-110	-57833	2520

Stresses - GASP Results

R	Z	σ_R	% DIFF	σ_Z	% DIFF	σ_T	% DIFF	σ_{RZ}	% DIFF
1.84	41.92	-53841	2.1	-10084	0	-53261	0.8	1910	2.3
1.82	41.42	-53903	1.2	-9126	1.0	-53755	0.7	2324	20.1
1.77	40.42	-54783	0.7	-6865	1.0	-54630	0.3	2524	21.2
1.73	39.42	-55811	0.3	-4367	0.9	-55652	0.1	2695	20.1
1.69	38.43	-56932	0.1	-1637	3.0	-56758	0.6	2826	16.6
1.66	37.93	-57501	0.4	-111	0.9	-57253	1.0	2530	0.4

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Table A.43-12

Shell Force Resultants

ELASTICITY SOLUTION	GASP SOLUTION			
Circumferential Stress	N_s	% DIFF	N_θ	DIFF
-220500	-221622	0.5	-220466	0.0

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Table A.43-13

Deflections

NODE	R	Z	ΔR ELASTICITY	ΔR GASP	% DIFF	ΔZ ELASTICITY	ΔR GASP	% DIFF
1	6.625	48.000	.00470375	.00470375	0.00	.1631419	.16301665	0.08
30	5.753	42.750	.00408463	.00408463	0.00	.1594144	.15928915	0.08
175	7.125	32.999	.00505875	.00505875	0.00	.1524912	.15236594	0.08
179	6.000	32.999	.00426000	.00426000	0.00	.1524912	.15236594	0.08
292	6.000	20.500	.00425997	.00426000	0.00	.1436169	.14349165	0.09
307	13.125	20.500	.00931873	.00931875	0.00	.1436169	.14349165	0.09
356	5.062	28.000	.00359402	.00359402	0.00	.1489419	.14881665	0.08
494	19.843	13.632	.01408848	.01408853	0.00	.1387408	.13861537	0.09
498	18.375	9.124	.01304640	.01304625	0.00	.1355401	.13541469	0.09
547	26.437	8.151	.01877058	.01877027	0.00	.1348498	.13472386	0.09
551	5.562	6.812	.00394902	.00394902	0.00	.1338984	.13377317	0.09
555	27.375	14.300	.01943606	.01943625	0.00	.1392156	.13908965	0.09

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A.43a GESOFIRE

The GESOFIRE digital computer code(1) solves a set of equations describing heat and mass transfer by the finite difference method. The physical system is simulated by a nodal network in which the nodes are connected to each other along a heat transfer path by admittances (the reciprocal of thermal resistance). Each node has a thermal capacity equal to that of the physcial counterpart represented by the node. The temperature assigned to or calculated by each node represents the temperature at the cetroid of the corresponding elements in the physical system.

Availability

The GESOFIRE code is available from General Electric in San Jose, California.

Verification

The GESOFIRE code[1] has been verified, i.e. the analytical expressions have been hand checked for correctness, and has been validated against small open air pool fire experiments performed at a sodium temperature of 1000°F. Validation against these experiments together with a present on-going validation effort for a large scale test (1000 lbs. of Na at 1000°F) will provide sufficient confidence in the accuracy of the code's sodium burning process.

Application

The GESOFIRE code is used to describe the pressure-temperature history in the RCB following a postulated sodium spill.

Reference

(1)General Electric Specification #23A2843, "THE GESOFIRE CODE."

A.44 GSAP4 (General Electric Proprietary)

GSAP4 is a proprietary computer code of the General Electric Company (GE). The program is the original version of SAP4GE but is on the CDC computer system. Refer to the program description for SAP4GE.

Availability

GSAP4 (October 1976) is available for GE use at Lawrence Berkeley Laboratory on the CDC 7600 computer and is maintained by GE-FBRD, Sunnyvale, California.

Verification

The validity of GSAP4 is well documented in Reference 1. A comparison of results is made in Figure A.34-1 per SRP Section 3.9.1.II.2.b.

Application

GSAP4 is applied to the static and dynamic analysis of the piping systems, dump tank, sodium pumps, etc.

Reference

Peterson, F. E., Bathe, K. J., Parker, C. S., Wassem, F. A., and Tang, Y. K., "SAP4GE-Static and Dynamic Analysis of Mechanical and Piping Components by Finite Element Method," NEDO-10909, August 1974. Engineering/Analysis Corporation prepared for General Electric Company.

A.45 HAA-3B

HAA-3B was developed at Atomics International for the analysis of the behavior of aerosols produced in sodium pool fire and other experiments there. It employs a time-dependent log-normal distribution model when agglomeration is prevalent, then switches to a stirred-setting model. Prediction-correction numerical integration techniques are used to solve a system of simultaneous, first-order differential equations. (Reference 1)

Availability

HAA-3B is available nationally from the Argonne Code Center.

Verification

Verification of the code results has been conducted against experimental measurements of the behavior of sodium oxide aerosols of low to moderate concentration in small to medium sized containment vessels, and of low concentration of uranium oxide aerosols. Further verification of HAA-3B has been conducted against validated computer programs. (References 2, 3, 4, 5 and 6)

Application

HAA-3B is regularly used to evaluate the aerosol behavior in containment for postulated accidents involving sodium, fission products, and fuel release.

References

(1) R. S. Hubner, E. V. Vaughan, L. Baurmash, "HAA-3 User Report," AI-AEC-13038, (3/73)

(2) L. Baurmash, R. P. Johnson, R. L. Koontz, C. T. Nelson, "Summary Report for Laboratory Experiments on Sodium Fires," TR-707-130-007 (8/73), Atomics International Report

(3) "Reassessment of AI Aerosol Modeling Efforts for Application to Hypothetical Reactor Accidents," Attachment to AI Letter 70AT-1955, H. A. Morewitz to A. J. Pressesky dated May 22, 1970

(4) "Analytic Studies of Aerosol Behavior Predictions for Fast Reactor Safety," Battelle Columbus Laboratories, BMI-1932, March 1975

(5) "Aerosol Behavior Modeling for Fast Reactor Safety," L. D. Reed, J. A. Gieseke, Battelle Columbus Laboratories, BMI-666, Quarterly Progress Report for July-September 1975, October 30, 1975

(6) NUREG/CR-1724, "Proceedings of the CSNI Specialists Meeting On Nuclear Aerosols in Reactor Safety, " October, 1980

A.46 HAFMAT

HAFMAT is a digital computer program written in FORTRAN language that calculates steady-state flow distribution through all paths of a given flow system. To accomplish this, the system is considered to be a network of flow paths or pipes. Flow geometry, heat additon, and system boundary conditions must be specified for every flow path within the system. Linear simultaneous equations are employed to insure a flow and thermal energy balance at each junction of two or more flows. Initially assumed flow distributions are modified by an iterative procedure until the correct flow distribution is obtained.

Availability

HAFMAT is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center. This code can be obtained from Argonne Code Center.

Verification

Verification according to SRP Section 3.9.1.II.2.c is discussed in the reference.

Application

HAFMAT is used to perform parametric studies of complex steady state flow systems for which heat addition to the system is a known function. It is used to analyze the CRBRP flow system during normal operation as well as at low flow rates, including natural circulation.

Reference

L. L. Wunderlich, D. R. Dolk, "HAFMAT, Steady-State Flow Distribution Program," KAPL-M-7128 (LXW-2), unpublished, Knolls Atomic Power Laboratory, General Electric Company, July, 1970.

A.47 HAP

The HAP program is an axisymmetric nonlinear heat analysis program. It uses finite element analogy to determine nodal temperatures in a two-dimensional or axisymmetric body subjected to transient disturbances.

The HAP program modified the finite element method as applied to heat transfer problems, and evaluates the temperature distribution within two-dimensional or axisymmetric bodies of arbitrary geometry. The nonlinearity effects of conduction, and cooling pipe forms of heat transfer are considered by the program.

Availability

The February, 1976 version of this program is available on the IBM 370, model 165 computer of the Chicago Bridge and Iron Company.

Verification

Verification of this program is achieved through the use of the following problems:

1. One Dimensional Steady State Condition:

   The problem used for this verification is ASME Benchmark Problem No. 4 of "Pressure Vessel and Piping 1972 Computer Programs Verification."

   The results obtained for this problem are tabulated in Table A.47-1.

2. Axisymmetric Thermal Transient Analysis with Forced Convection:

   The verification problem presented is ASME Benchmark Problem No. 10 of "Pressure Vessel and Piping 1972 Computer Programs Verification".

   The geometry and convection boundaries are shown in Figure A.47-1. The results obtained with HAP compare favorably with those given by the ASME in the "Pressure Vessel and Piping 1972 Computer Programs Verification', and those calculated by TAP-A in "Verification of Heat Transfer Computer Code 'TAP-A'" by W. N.Davis, E. H. Novendstern, LRS-75-0789, August 1975 and WECAN in "Problems for Verification of Heat Transfer Computer Code 'WECAN'" by J.J. Buggy WG50252. Figures A.47-2, A.47-3 and A.47-4 show the results of temperature as a function of radius for several transient times at the three locations A, B, & C indicated in Figure A.47-1.

3. One and Two Dimensional Transient Heat Conduction Analysis in a Square Slab:

   The verification problem presented is ASME Benchmark Problem No. 28 of "Pressure Vessel and Piping 1972 Computer Programs Verification."

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HAP cont'd.

The problem consists of a 10" square slab initially at 100°F suddenly exposed to a fixed temperature boundary of 0°F. Two cases are analyzed. Case one has three sides insulated, and the fourth side held at 0°F. The second case has two adjacent sides insulated, with the remaining two sides held at 0°F. The results of this problem are as follows:

For both cases, three different time increments were used; 10, 1, and .1 seconds. The temperature histories of the cover furthest from the quenched edges are shown for the two cases in Figure A.47-5 and A.47-6 respectively. The comparisons between temperatures when the number of interactions is changed are given in tables A.47-2 and A.47-3. The exact solutions were obtained from V.S. Arpaci "Conduction Heat Transfer" Addison-Wesley (1966) page 292.

4. Temperature Response of a Plate Exposed to Constant Temperature Radiation:

The problem was solved twice using the two radiation features of the HAP program. For case 1 the radiation boundary condition option of the HAP program is used. For Case 2 a radiation enclosure is used. The geometry and boundary conditions for the plate are shown in Figures A.47-7 and A.47-8 for Cases 1 and 2 respectively.

The temperature results for Case 1 are shown in Table A.47-4 and the results of Case 2 are shown in Table A.47-5.

5. Steady State Conduction in Plate Composed of Two Materials:

The problem considered is a square plate with two opposite boundary conditions at 100°F and the other two boundaries at 0°F and 200°F. This plate is divided into two equal rectangles of conductivity ratio 2:1.

The exact solution for this problem was obtained from "Finite-Difference Methods for Inhomogenous Regions," Trezek, G. J., and J. G. Witwer, "Transactions of the ASME Journal of Heat Transfer," August 1972, Pages 321-323, and the HAP results showed good comparison with a maximum error of 2.14%. The temperature results for selected nodes are tabulated in Figure A.47-8.

6. Two-dimensional Response of a Cylinder Caused by Internal Heat Generation:

The problem considered is that of an insulated cylinder having an internal heat generation history. This cylinder has an inside radius of 10 feet with a cross-sectional area of 4 square feet. Figure A.47-9 shows the geometry and heat flow volume of the cylinder. The input heat generation time history is shown in Figure A.47-10.

The results obtained with HAP are in good agreement with those of TAP-A and WECAN referenced in 1 above and ANSYS-C.J. Desalvo and J.S. Swanson, ANSYS Engr. Analysis System, Swanson Analysis System , Inc., Elizabeth, PA, 1975. The results at three locations in the cylinder are shown in Figure A.47-11 with ANSYS, WECAN and TAP-A results.

7. Transient Heat Conduction in a Pipe with Convection Boundary Conditions:

The HAP solution is compared to a solution obtained with TAP-A program referenced in problem 1 above and the "MARC" computer program "MARC-CDC Users Information Manual Volume III Demonstration Problems "MARC Analysis Corporation, and Control Data Corp.

The problem considered is that of a long cylinder, initially at 1100°F, subjected to instantaneous convection boundaries at the inside and outside surface with free stream temperature of 800°F.

The results are compared in Figure A.47-12.

8. One Dimensional Radiation Through a Gray Cylinder:

The problem consists of radiation heat transfer through a long cylindrical hole, having gray adiabatic walls. The problem is run out to the steady state condition and the HAP results are compared to an analytical solution.

The cylinder has an inside radius of 5 inches, is 0.25 inches thick and 19 inches tall. The problem geometry and finite element model is shown in Figure A.47-13. A radiation enclosure is used inside the cylinder as shown in Figure A.47-14. The cylinder was arbitrarily given a uniform temperature of 1000°F for the initial temperature. The HAP result for the heat transfer Q at the top surface equal to 4,587.18 BTU/HR which is a 4.65% error.

9. Radiant Heat Transfer.

Five different cases were run with different temperatures assigned to the base of the fin. Each case was run out to steady state and the net heat transfer rate for each enclosure was recorded for comparison with an analytical solution. The problem geometry and finite element model is shown in Figure A.44-15.

From the reference "Radiant Heat Transfer from a Flat Plate Uniformly Heated on One Edge," D. B. MacKay, and E. L. Leventhal, Advanced Engineering, North American Aviation, Inc. Missile Development Division, August 13, 1958, MD58-187. - the heat transferred away from the plate by radiation is given by equation (18) of the above reference. Table A.47-4 compares the heat transfer rates calculated by HAP and those calculated by the equation referenced above. In the above equation on the right hand side, the only unkown is Tc. This value was taken from the HAP solution for use in the formula. As shown in Table A.47-6, HAP accurately predicts the net heat tranfer rate from the fin for the range of T_h values.

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Application

HAP will be used in the Thermal Analysis of the reactor vessel closure head assembly and its components.

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Table A.47-1

Radius (in.)	TEMPERATURE			HAP % Dev. of Theory
	Theory	TAP-A	HAP	
10.25	93.91	93.90	94.0	.096
10.75	82.16	82.14	82.4	.292
11.25	70.95	70.92	71.3	.493
11.75	60.23	60.21	60.6	.614
12.25	49.95	49.94	50.4	.901
12.75	40.08	40.08	40.5	1.048
13.25	30.60	30.60	31.0	1.307
13.75	21.46	21.46	21.8	1.584
14.25	12.65	12.65	12.8	1.186
14.75	4.15	4.15	4.2	1.205

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Table A.47-2

One Dimensional Heat Conduction

TIME (sec)	TEMPERATURE AT POINT A (°F)						
	EXACT	HAP ΔT = 10 sec.	% ERROR	HAP ΔT = 1 sec.	% ERROR	HAP ΔT = .1 sec.	% ERROR
10	94.93	91.4	3.72	94.2	0.77	94.8	0.14
20	77.23	77.2	0.04	77.0	0.30	76.9	0.43
30	60.68	63.2	4.15	60.5	0.30	60.1	0.96
40	47.45	50.9	7.27	47.2	0.53	46.8	1.37
50	37.08	40.8	10.03	36.9	0.49	36.4	1.83
60	28.97	32.7	12.88	28.7	0.93	28.3	2.31
70	22.64	26.1	15.28	22.4	1.06	22.0	2.83
80	17.69	20.9	18.15	17.5	1.07	17.1	3.34
90	13.82	16.7	20.84	13.6	1.59	13.3	3.76
100	10.80	13.3	23.15	10.6	1.85	10.3	4.63

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Table A.47-3

Two Dimensional Heat Conduction

TIME (sec)	TEMPERATURE AT POINT A (°F)						
	EXACT	HAP ΔT = 10 sec.	% ERROR	HAP ΔT = 1 sec.	% ERROR	HAP ΔT = .1 sec	% ERROR
10	90.12	86.0	4.57	90.0	0.13	90.9	0.87
20	59.64	64.8	8.65	61.7	3.45	61.2	2.62
30	36.82	45.9	24.66	39.0	5.92	38.0	3.20
40	22.52	31.6	40.32	24.3	7.90	23.4	3.91
50	13.75	21.5	56.36	15.1	9.82	14.3	4.0
60	8.39	14.5	72.82	9.3	10.85	8.8	4.89
70	5.13	9.7	89.08	5.8	13.06	5.4	5.26
80	3.13	6.5	107.67	3.6	15.02	3.3	5.43
90	1.91	4.4	130.37	2.2	15.18	2.0	4.71
100	1.17	2.9	147.86	1.4	19.66	1.2	2.56

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Table A.47-4

Case 1 Temperature Results

TIME (sec)	$T_{(ref\ 1.5)}$	$T_{(Run\ 1)}$	% Difference	$T_{(Run\ 2)}$	% Difference
2	400°R	416.39	3.9	404.59	1.1
4	378°R	390.89	3.3	384.19	1.6
10	340°R	351.69	3.3	348.19	2.4
20	300°R	316.29	5.2	308.29	2.7
40	253°R	268.69	5.8	258.69	2.2
100	193°R	200.69	3.8	193.79	0.4
200	153°R	165.49	7.5	153.99	0.6
400	120°R	130.39	8.0	121.29	1.1
1000	90°R	92.79	3.0	88.49	1.7
2000	70°R	76.09	8.0	70.29	0.4
4000	55°R	59.89	8.2	55.59	1.1

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Table A.47-5

Case 2 Temperature Results

TIME (sec)	$T_{(ref\ 1.5)}$	$T_{(run\ 1)}$	% Difference	$T_{(run\ 2)}$	% Difference
2	400°R	419.69	4.7	407.19	1.8
4	378°R	394.49	4.2	386.89	2.3
10	340°R	354.59	4.1	351.19	3.2
20	300°R	319.59	6.1	311.49	3.7
40	253°R	271.59	6.8	261.99	3.4
100	193°R	203.29	5.1	196.69	1.9
200	153°R	168.09	9.0	156.49	2.2
400	120°R	132.39	9.4	123.39	2.7
1000	90°R	94.19	4.4	89.99	0.0
2000	70°R	77.49	9.7	71.59	2.2
4000	55°R	60.79	9.5	56.59	2.8

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Table A.47-6

T_h (^oF)	T_h (^oR)	T_c (^oR)	T_c/T_h	Q Analytical	Q HAP	% Error
1200	1660	1173.5	0.707	22,912,53	22,844.5	0.297
1000	1460	1101.6	0.755	15,913.73	15,887.1	0.167
800	1260	1014.8	0.8055	10,303.93	10,290.38	0.1315
600	1060	909.3	0.8578	6,021.1	6,014.9	0.103
400	860	781.3	0.9085	3,011.4	3,010.44	0.032

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Geometry and Boundary Conditions:

Units: Length - inches
Film Coefficient - Btu/min.-°F-in.^{2}

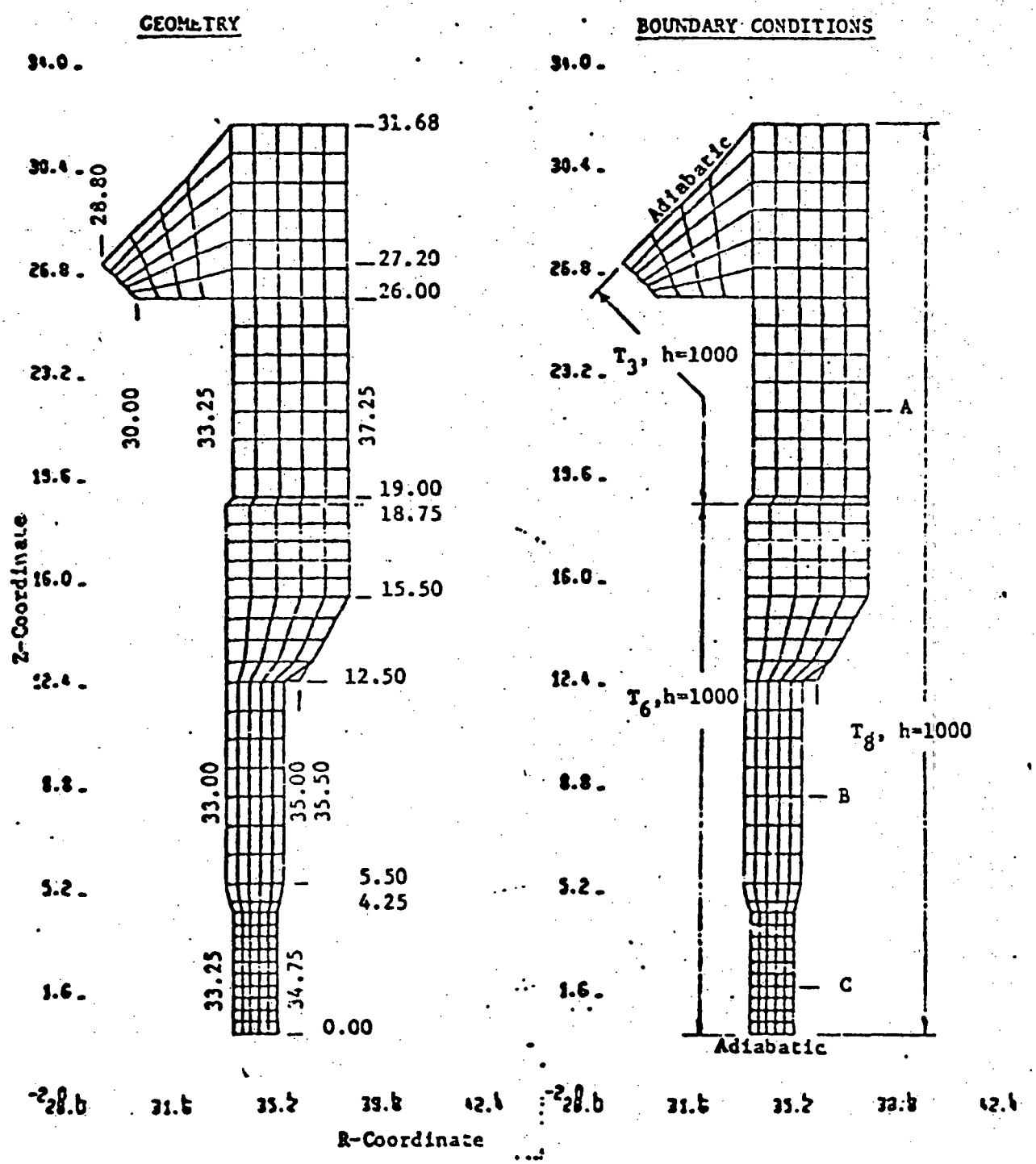

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Figure A.47-1

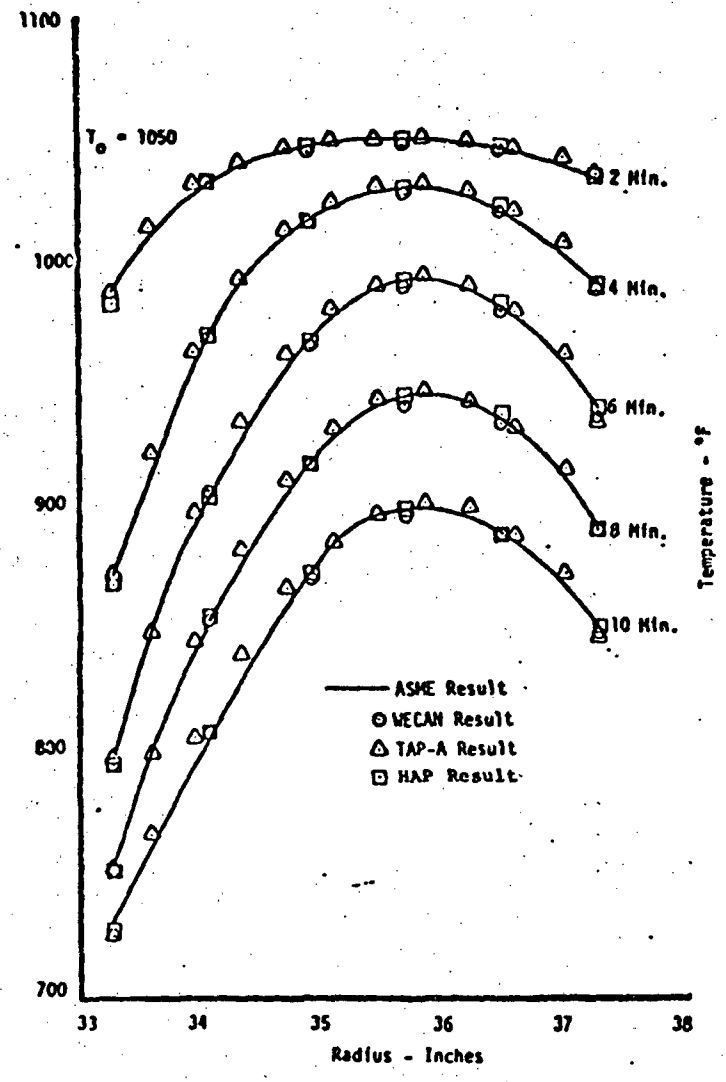

Figure A.47-2

TEMPERATURE DISTRIBUTION AT SECTION A

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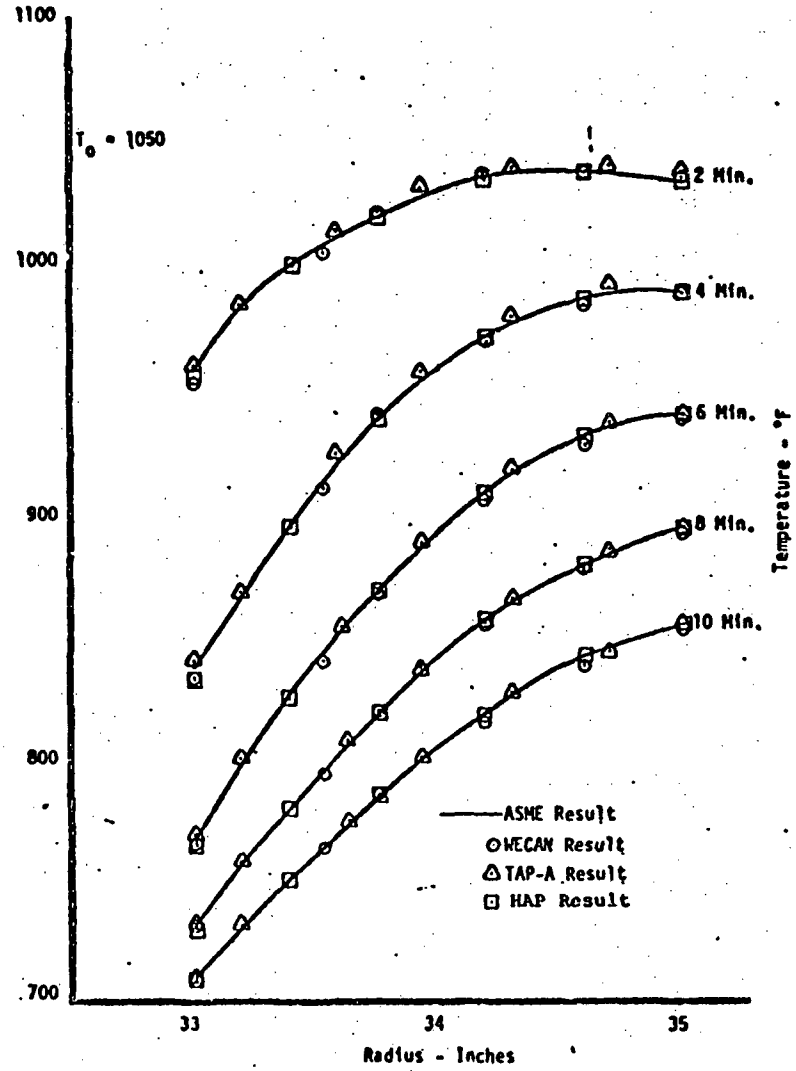

Figure A.47-3
TEMPERATURE DISTRIBUTION AT SECTION B

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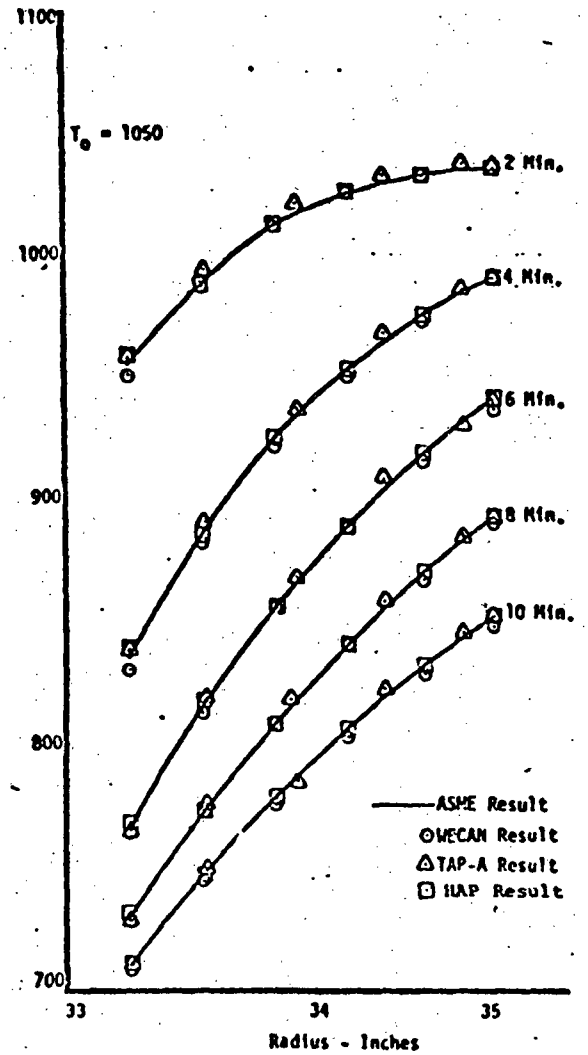

Figure A.47-4
TEMPERATURE DISTRIBUTION AT SECTION C

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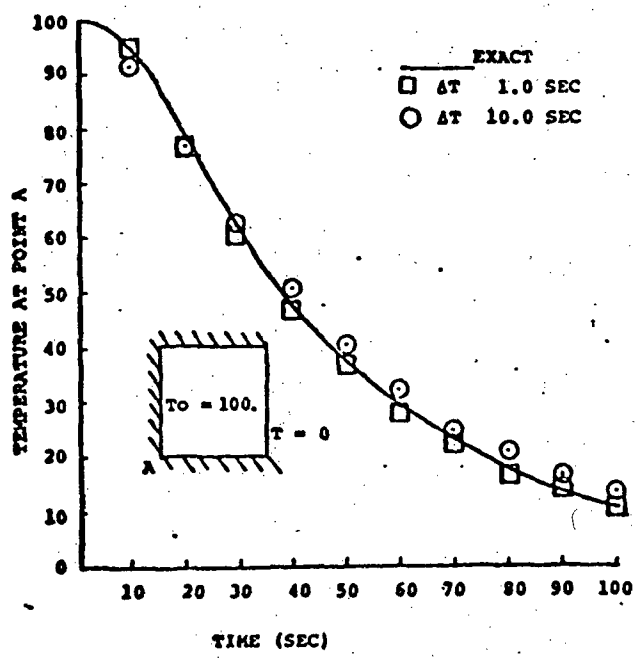

One Dimensional Heat Conduction in Square Plate

Figure A.47-5

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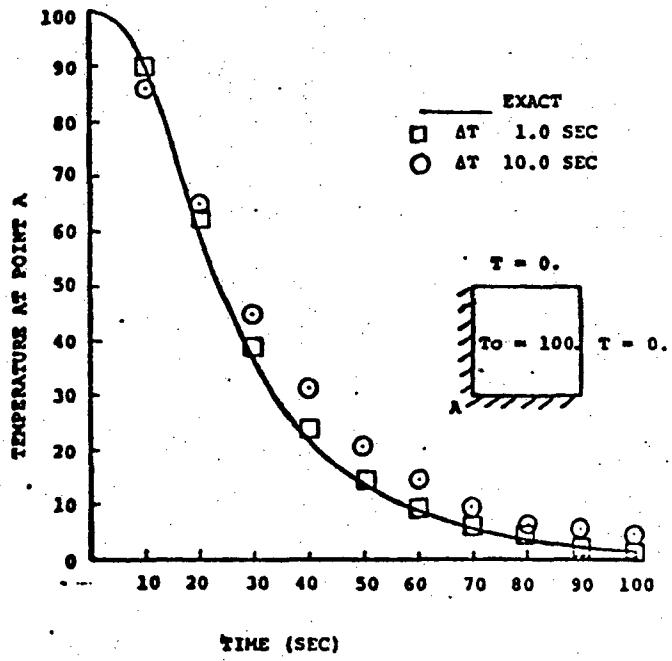

Two Dimensional Heat Conduction in Square Plate

Figure A.47-6

Amend. 45
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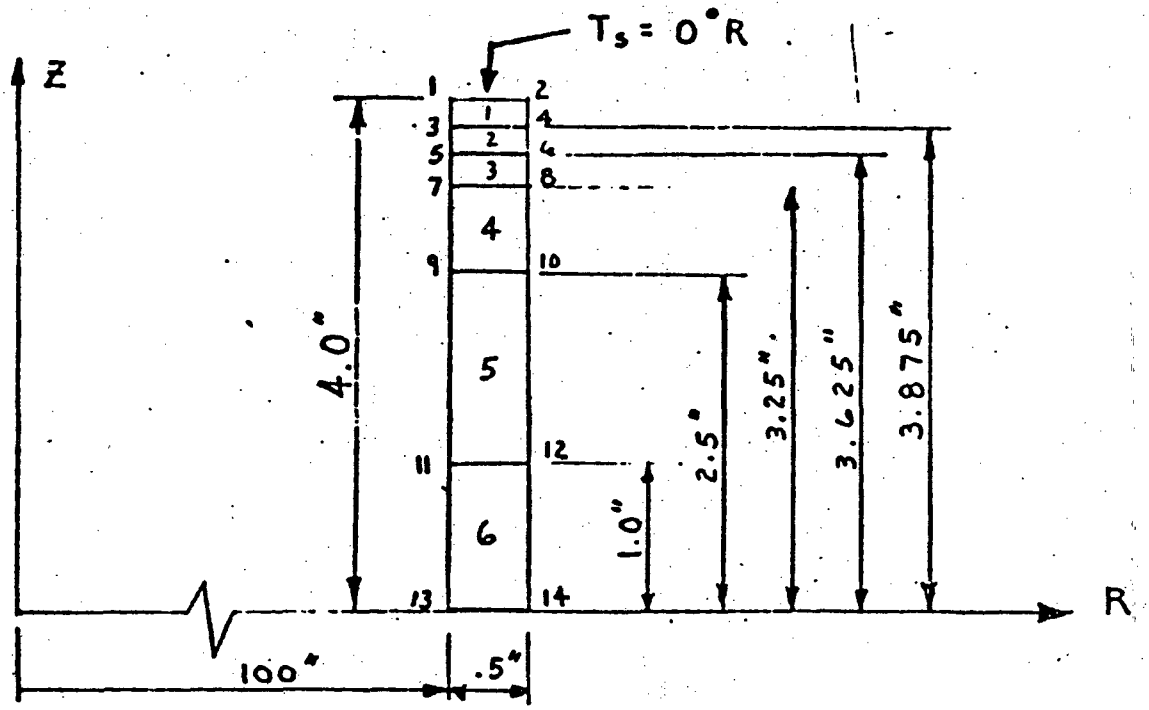

FIGURE A.47-7

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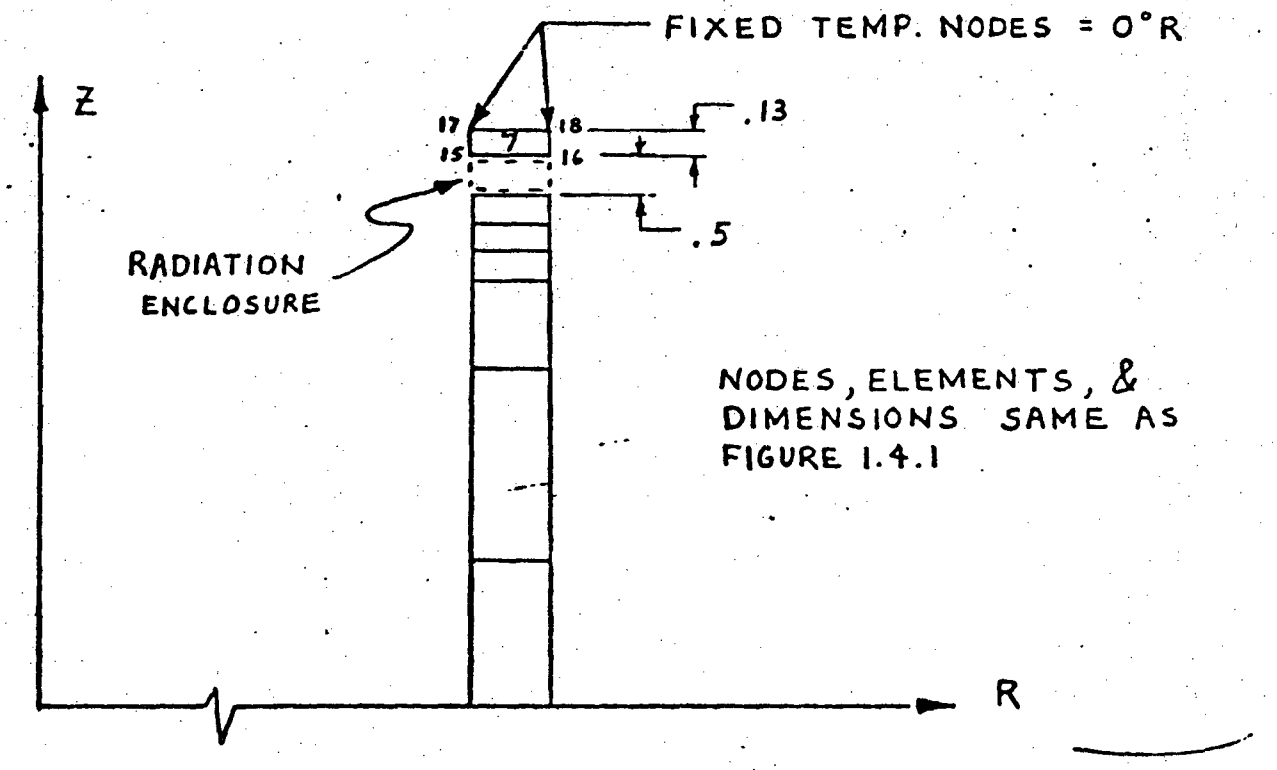

FIGURE A.47-8

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147.58 147.5 0.05	165.50 166.2 0.42	172.44 172.8 0.21	175.34 175.5 0.09	176.43 176.6 0.10
124.83 124.1 0.58	140.35 140.3 0.04	148.44 148.6 0.11	152.32 152.5 0.12	153.87 154.0 0.08
113.01 112.6 0.36	122.63 122.4 0.19	128.45 128.4 0.04	131.50 131.6 0.08	132.77 132.9 0.10
105.03 104.8 0.22	108.86 108.7 0.15	111.21 111.1 0.10	112.40 112.4 0.00	112.86 112.9 0.04
89.07 89.2 0.15	79.82 80.0 0.23	73.07 73.1 0.04	68.81 68.8 0.01	66.78 66.7 0.12
82.96 83.3 0.41	69.64 69.8 0.23	60.79 60.8 0.02	55.60 55.5 0.18	53.23 53.1 0.24
72.66 73.3 0.88	54.83 54.8 0.05	44.85 44.6 0.56	39.62 39.4 0.56	37.38 37.2 0.48
51.21 51.2 0.02	32.19 31.5 2.14	24.34 24.0 1.40	20.79 20.6 0.91	19.36 19.2 0.83

— EXACT
— HAP
— % ERROR

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FIGURE A.47-9

TEMPERATURE RESULTS

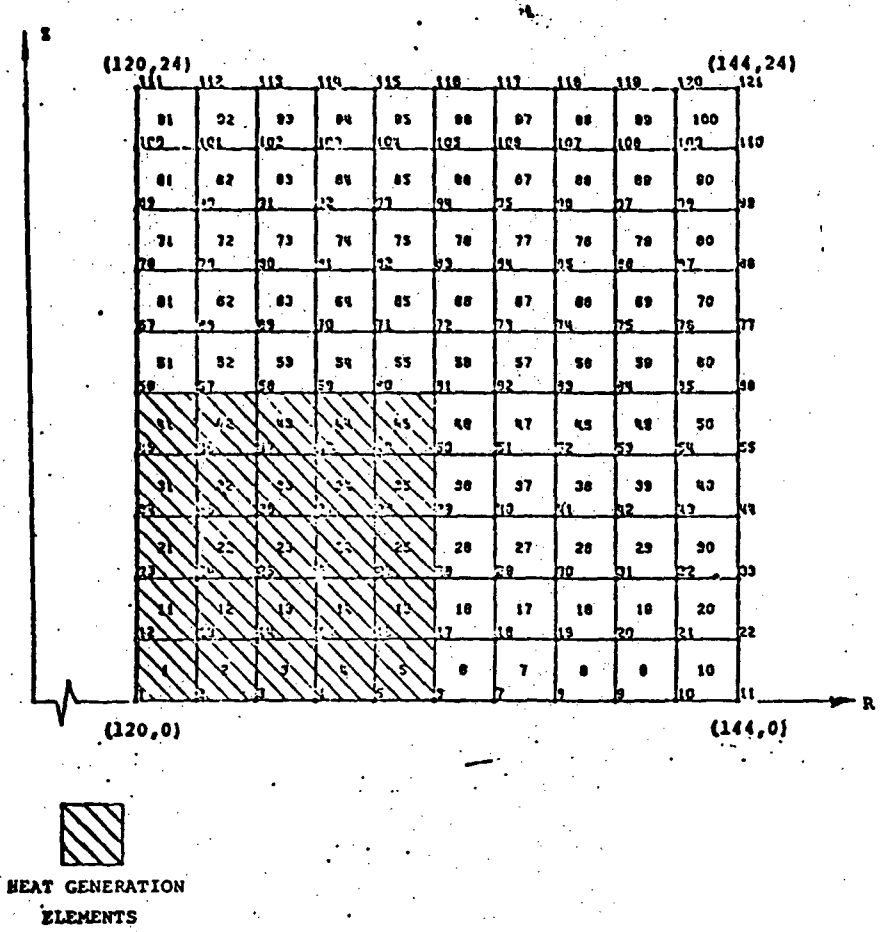

FIGURE A.47-10
CONFIGURATION AND GEOMETRY

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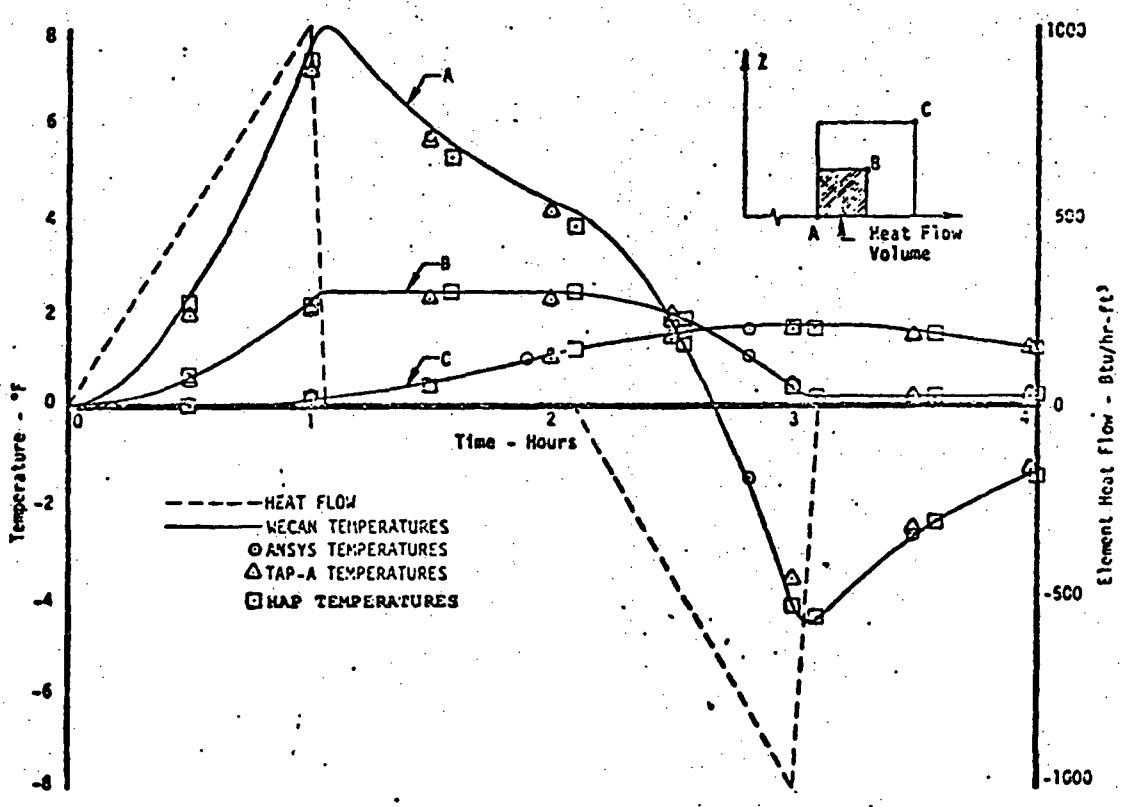

FIGURE A.47-11

TRANSIENT TEMPERATURE DISTRIBUTION

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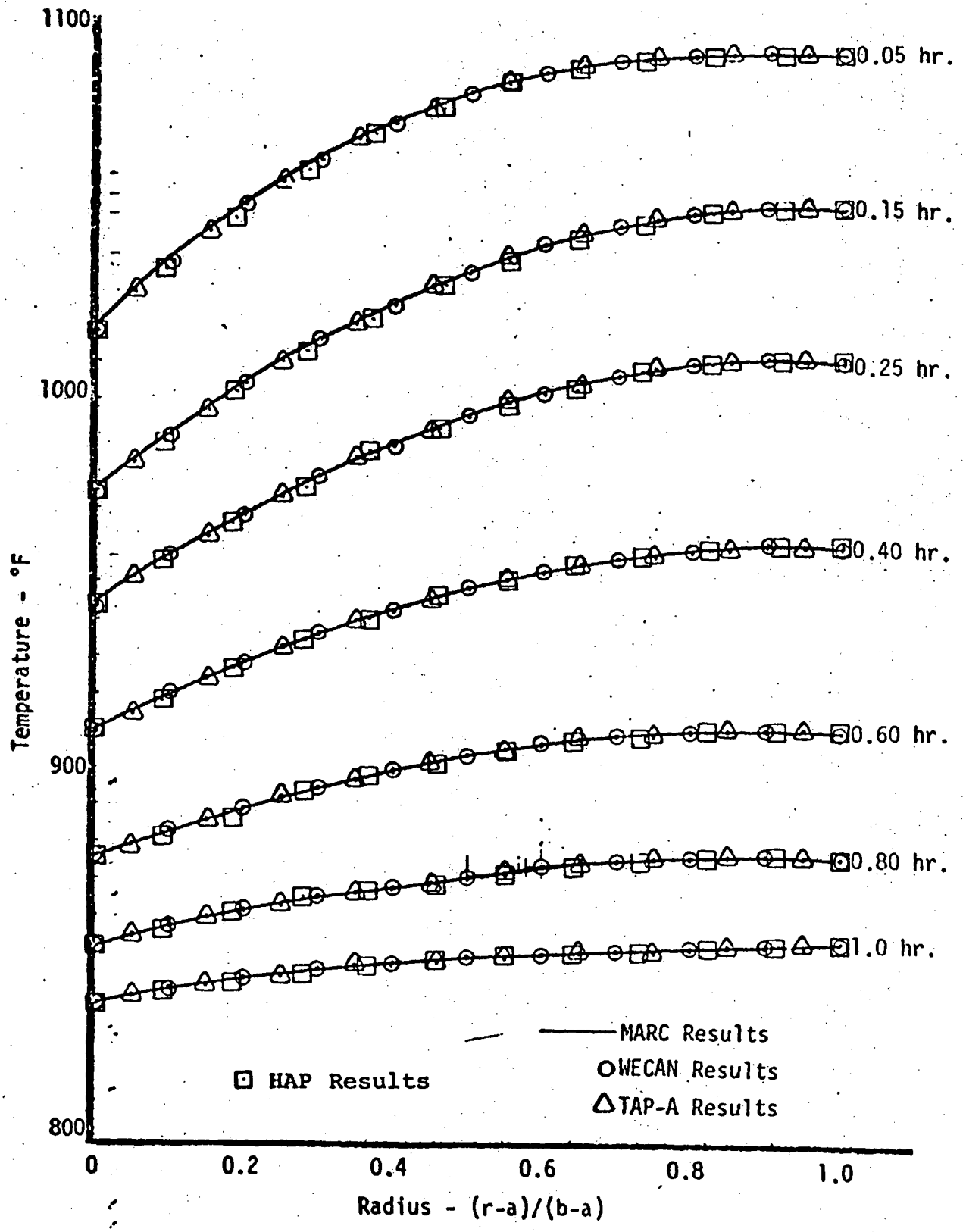

FIGURE A.47-12

TRANSIENT TEMPERATURE DISTRIBUTION

Amend. 45
July 1978

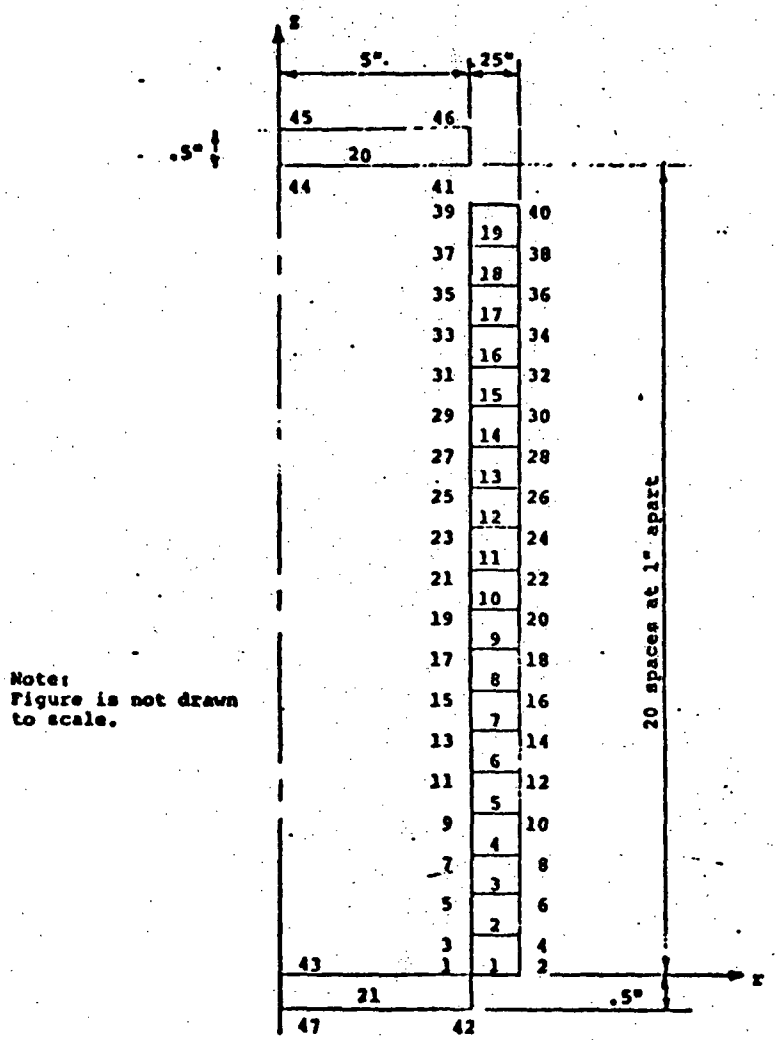

FIGURE A.47-13

PROBLEM GEOMETRY AND FINITE ELEMENT MODEL

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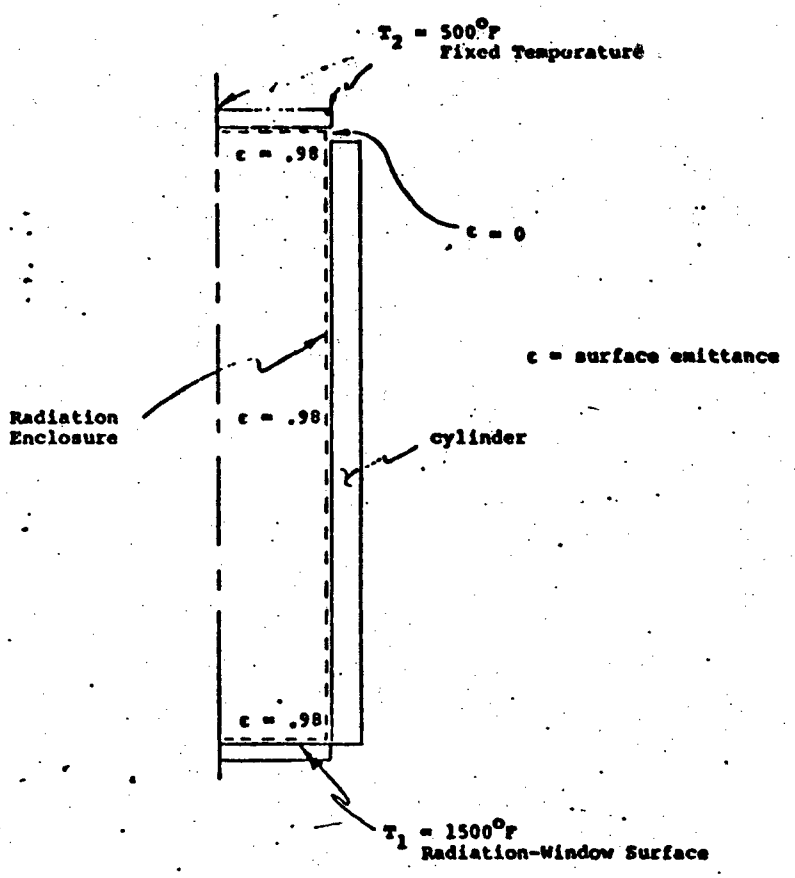

FIGURE A.47-14

RADIATION ENCLOSURE AND BOUNDARY CONDITIONS

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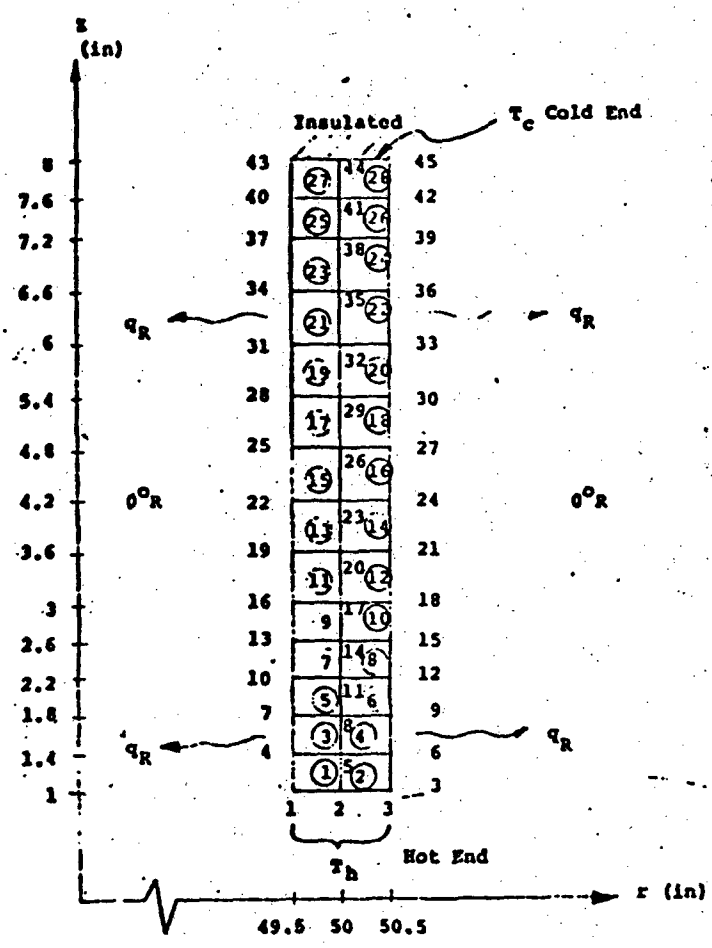

FIGURE A.47-15
PROBLEM GEOMETRY AND FINITE ELEMENT MODEL

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A.48 HAP-II (SAP) General Electric Proprietary)

HAP II (SAP), a proprietary computer program of the General Electric Company, is a two-dimensional, axisymmetric, nonlinear transient temperature analysis computer code with external linkage to SAP IV for determining thermal stresses. An unconditionally stable numerical integration scheme is combined with an iteration procedure to compute temperature distributions.

Arbitrary geometry can be modeled. Time and temperature dependent boundary conditions (radiation, forced, and free convection and fluid flow) are permitted.

Geometry plots for data checking are also available.

Availability

HAP II (SAP) has been available for GE use on the Honeywell 6000 of Nuclear Energy System Division, General Electric Company, San Jose, California since May, 1974.

Verification

HAP II (SAP) has been well demonstrated by solving a series of bench mark problems to produce accurate results as compared with the results from analytical approaches and ANSYS (A.3). See Figure A.48-1.

Application

HAP II (SAP) can be applied to any two dimensional and axisymmetric structures. Its main usage will be in the thermal analysis of the various piping systems.

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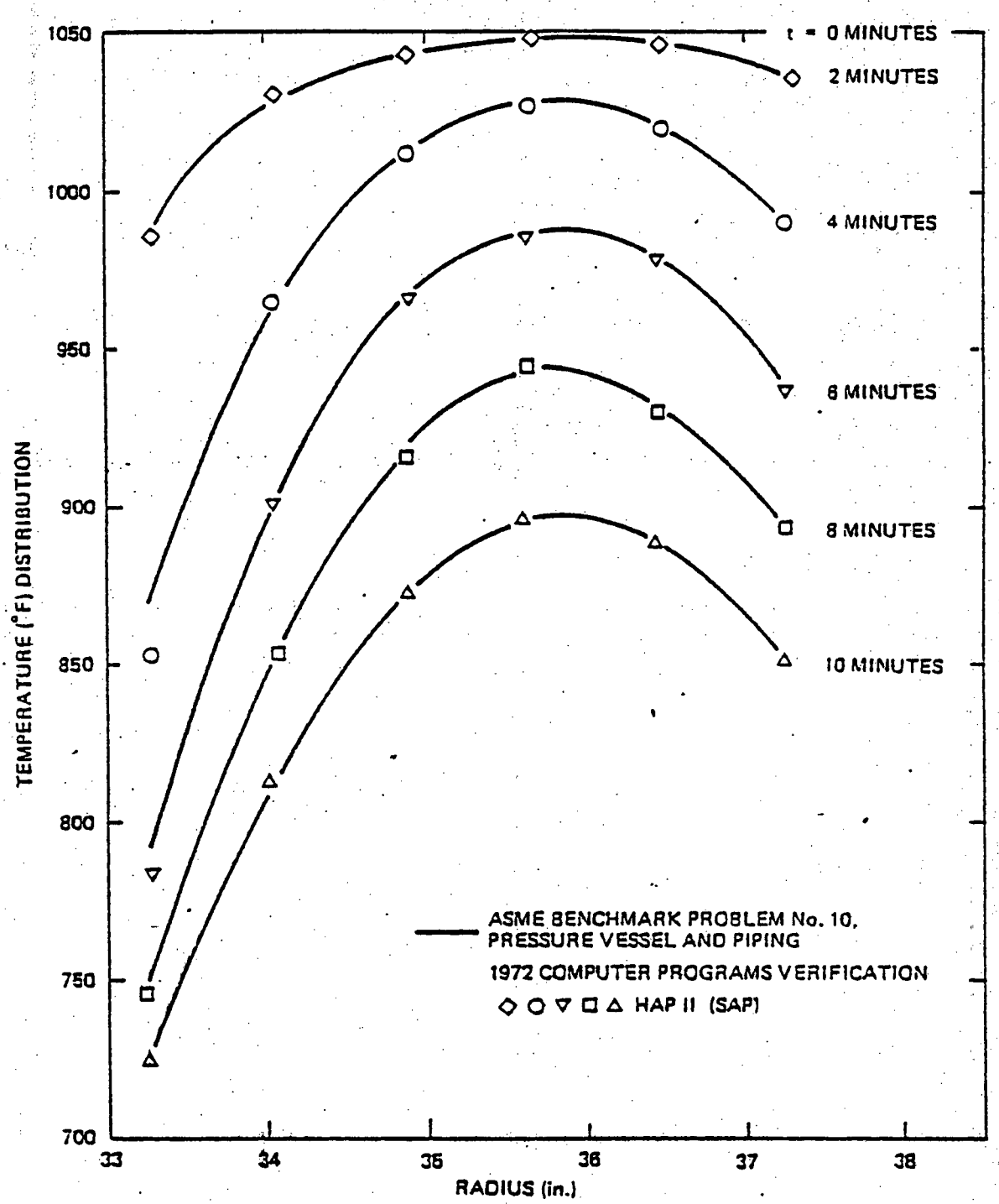

Figure A.48-1 *Comparison of HAP-II and Benchmark Problems of the ASME*

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HETHA AND VETHA

HETHA and VETHA are computer programs designed to perform the seismic analysis of structures including the effects of soil-structure interaction.

Based on a lumped-mass representation of the structure and the soil, the program solves the coupled equations of motion by direct integration, allowing for a rigorous solution of the soil-structure interaction problem. The formation of the equations of motion is similar to that used by Tsai in Reference 1.

This approach considers the displacements of the nodes of the structure subjected to an earthquake as composed of two parts: The displacements relative to the base which are expressed in terms of the modes of vibration of the "fixed" base structure and the displacements due to the base motion. The equations of motion are expressed in terms of the mode shapes, modal frequencies and modal dampings of the "fixed" base structure and of the stiffness and damping coefficents of the lumped springs and dashpots that represent the effects of the soil on the structure.

As a result of this formulation the number of coupled equations to be integrated is reduced from N+P to M+P, where N is the number of degrees of freedom of the "fixed" base structure, P the number of degrees of freedom of the base and M the number of modes of the "fixed" base structure selected to represent the structure. For a large structure M is much less than N.

While Tsai's formulation includes only two degrees of freedom (horizontal translation and rocking) the HETHA formulation considers three degrees of freedom per node (horizontal translation, rocking and torsion). VETHA considers one degree of freedom per node (vertical translation). In Ref. 1, Tsai proposed the use of the exact formulation to calculate equivalent modal dampings for a modal analysis of the structures by matching the responses of the exact analysis with those of a modal analysis. In HETHA and VETHA, the coupled equations of motion, similar to equation (15) of Ref. 1 are integrated directly using Wilson's step by step integration method and the acceleration time-history of the ground motion as input.

Availability

HETHA and VETHA are computer programs developed by Burns and Roe and are being used with CDC-7600 computers.

Verification

Results from HETHA and VETHA have been checked against the results obtained for the same problems with the ANSYS computer program. ANSYS is a recognized and accepted computer program. The verification follows.

Verification of HETHA

Two similar lumped-mass mathematical models with foundation springs and dampers were run with the computer programs HETHA and ANSYS as a verification of HETHA. The mathematical models consist of three mass points connected by flexible members. Three degrees of freedom (horizontal displacement, rotation about a horizontal axis perpendicular to the displacement and torsion) were allowed at each node. Foundation springs and dashpots for horizontal translation, rocking and torsion at the base of the model represent the soil-structure interaction effect. The mathematical model is shown in Figure A.49-1. In the HETHA model the structural damping is represented by the damping ratio of the "fixed" base structure (7% of critical). For the ANSYS model, based on the 7% damping ratio of the "fixed" base structure, the damping coefficients for the structural members were calculated. They are shown in Figure

Using the acceleration time-history for a horizontal ground motion, the responses were calculated using both HETHA and ANSYS. Both computer programs use direct integration of the coupled equations of motion.

The responses from both calculations show good agreement.

Table A.49-1 shows a comparison of maximum displacements and the time at which they occurred. Figures A.49-3, 4 and 5 compare the acceleration response spectra calculated with each program for the three degrees of freedom of the upper mass point (Node 1). Similar agreement was obtained at the other mass points.

Verification of VETHA

The verification of VETHA is similar to the verification of HETHA. The responses of a mathematical model with one degree of freedom at each mass point (vertical translation) were calculated using the computer programs VETHA and ANSYS. An acceleration time-history for a vertical ground motion was used as input.

Figure A.49-6 shows the mathematical model. Figure A.49-7, the damping coefficients used in the ANSVS input for the structural members. They correspond to a damping ratio of 7% for the "fixed" base structure. The results from both calculations show good agreement.

Table A.49-2 shows a comparison of maximum displacements and the time at which they occurred. Figure A.49-8 compares the acceleration response spectra calculated with each program for the upper mass point of the mathematical model (Node 1). Similar agreement was obtained at the other mass points.

TABLE A.49-1

COMPARISON OF COMPUTER PROGRAMS
HETHA AND ANSYS

MAXIMUM DISPLACEMENTS

NODE	TYPE	ANSYS		HETHA		% RATIO
		VALUE	TIME	VALUE	TIME	$\frac{\text{ANSYS}}{\text{HEATHA}} \times 100$
1	Translation	-1.89×10^{-4}	1.45	-1.93×10^{-4}	1.45	97.9
2	Translation	-9.27×10^{-5}	1.46	-9.7×10^{-5}	1.46	95.6
3	Translation	-6.57×10^{-5}	1.46	-6.7×10^{-5}	1.46	98.1
1	Rotation	-6.75×10^{-7}	1.45	-6.9×10^{-7}	1.45	97.8
2	Rotation	-6.31×10^{-7}	1.45	-6.5×10^{-7}	1.45	97.8
3	Rotation	-4.91×10^{-7}	1.45	-5.0×10^{-7}	1.45	98.2
1	Torsion	4.34×10^{-6}	1.96	-4.34×10^{-6}	1.96	100.0
2	Torsion	4.35×10^{-6}	1.96	-4.38×10^{-6}	1.96	99.3
3	Torsion	4.32×10^{-6}	1.96	-4.36×10^{-6}	1.96	99.1

NOTE:

Units

Displacements: ft.

Time: seconds

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TABLE A.49-2

COMPARISON OF COMPUTER PROGRAMS
VETHA AND ANSYS

MAXIMUM DISPLACEMENTS

NODE	ANSYS		VETHA		% RATIO
	VALUE	TIME	VALUE	TIME	$\frac{\text{ANSYS}}{\text{VETHA}} \times 100$
1	1.445×10^{-4}	1.45	1.51×10^{-4}	1.45	95.7
2	1.072×10^{-4}	1.45	1.12×10^{-4}	1.45	95.7
3	0.667×10^{-4}	1.46	0.69×10^{-4}	1.46	96.7

NOTE:

Units:

Displacements: ft.

Time: seconds

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TABLE A. 49-3

Symbols and Units

M - Mass (Kip sec^2/ft)

Mzz - Mass moment of inertia about axis Z (Kip sec^2 ft)

M_j - Mass moment of inertia about axis x (Kip sec^2 ft)

e - Eccentricity of mass point with respect to center of rigidity of member (ft)

H - Elevation of mass point (ft)

As - Shear area of member (ft^2)

Izz - Moment of inertia of member about axis Z (ft^4)

J - Torsional constant of member (ft^4)

Ky - Soil spring constant for translation in the Y direction (Kip/ft)

Kxx - Soil spring constant for torsion (Kip ft)

Kzz - Soil spring constant for rocking (kip ft)

Cy - Damping coefficient for Horizontal translation (Kip sec/ft)

Cθ - Damping coefficient for rotation (Kip ft sec)

Cø - Damping coefficient for torsion (Kip ft sec)

Cθy - Damping coefficient for coupled translation and rotation (Kip sec)

A - Area of member (ft^2)

Kx - Soil spring constant for vertical translation (Kip/ft)

C - Damping coefficient for vertical translation (Kip sec/ft)

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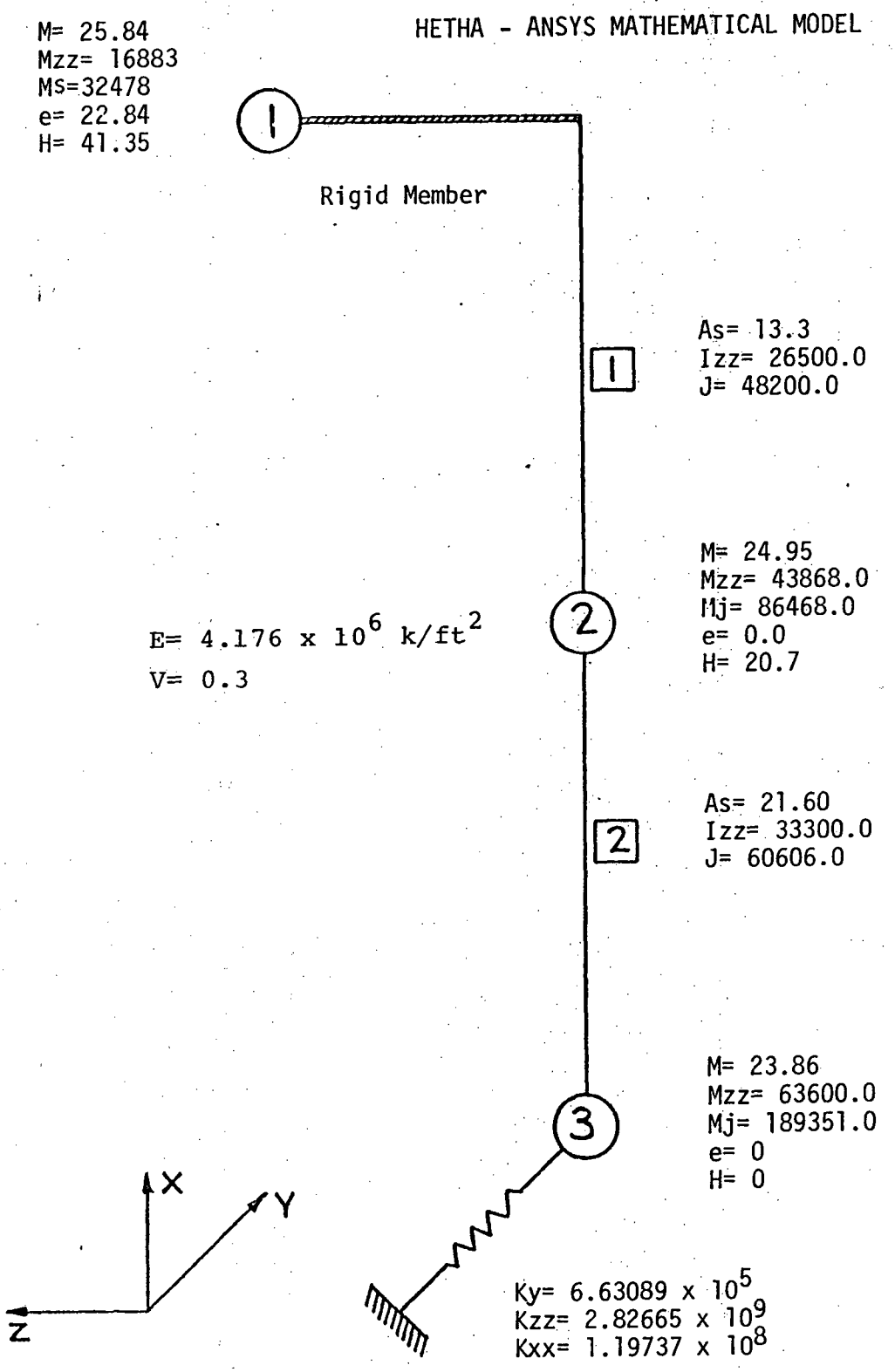
HETHA - ANSYS MATHEMATICAL MODEL



Notes:

1- For damping values see Figure A.49-2

2- For symbols and units see Table A. 49-3

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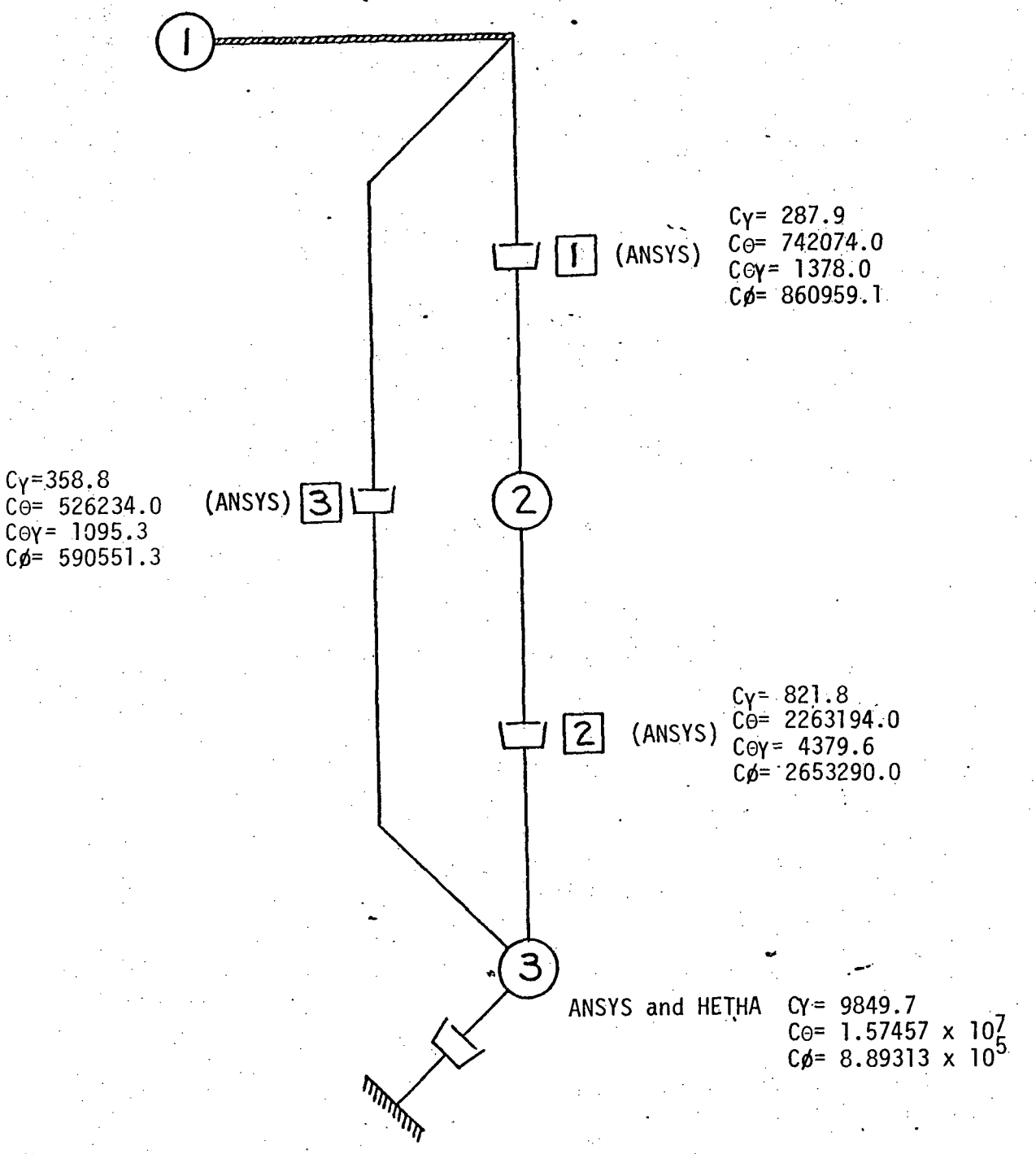
FIGURE A.49-1

HETHA - ANSYS-MATHEMATICAL MODEL-DAMPING VALUES



Notes:

1- Structural damping for HETHA: 7% of critical based on a "fixed" base condition
2- For stiffness and mass properties of model, see Figure A.49-1
3- For symbols and units see Table A.49-3

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FIGURE A.49-2

HETHA - ANSYS COMPARISON

NODE 1 - HORIZONTAL TRANSLATION

ACCELERATION RESPONSE SPECTRA - 3% DAMPING

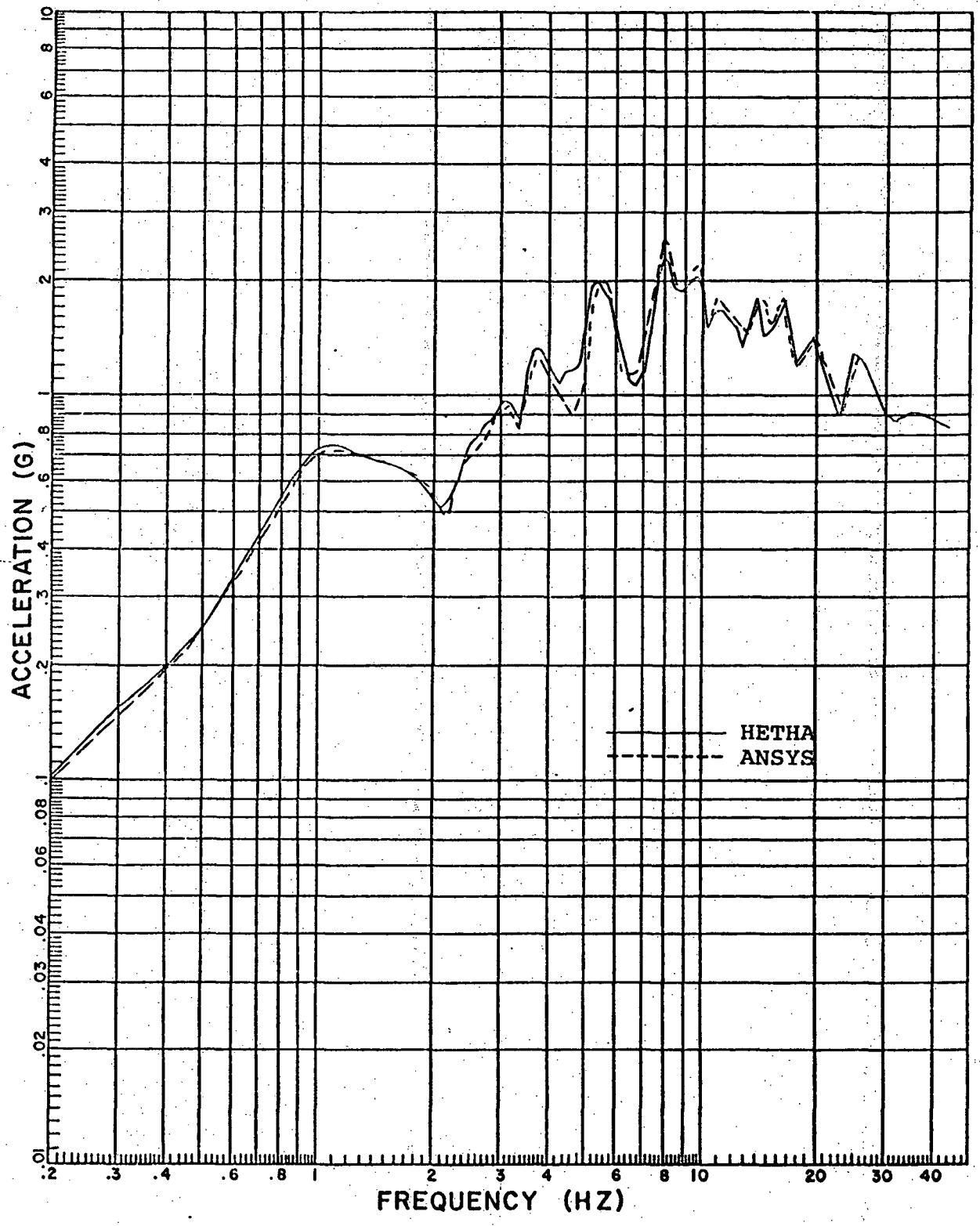

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FIGURE A.49-3

HETHA - ANSYS COMPARISON

NODE 1 - ROTATION

ACCELERATION RESPONSE SPECTRA - 3% DAMPING

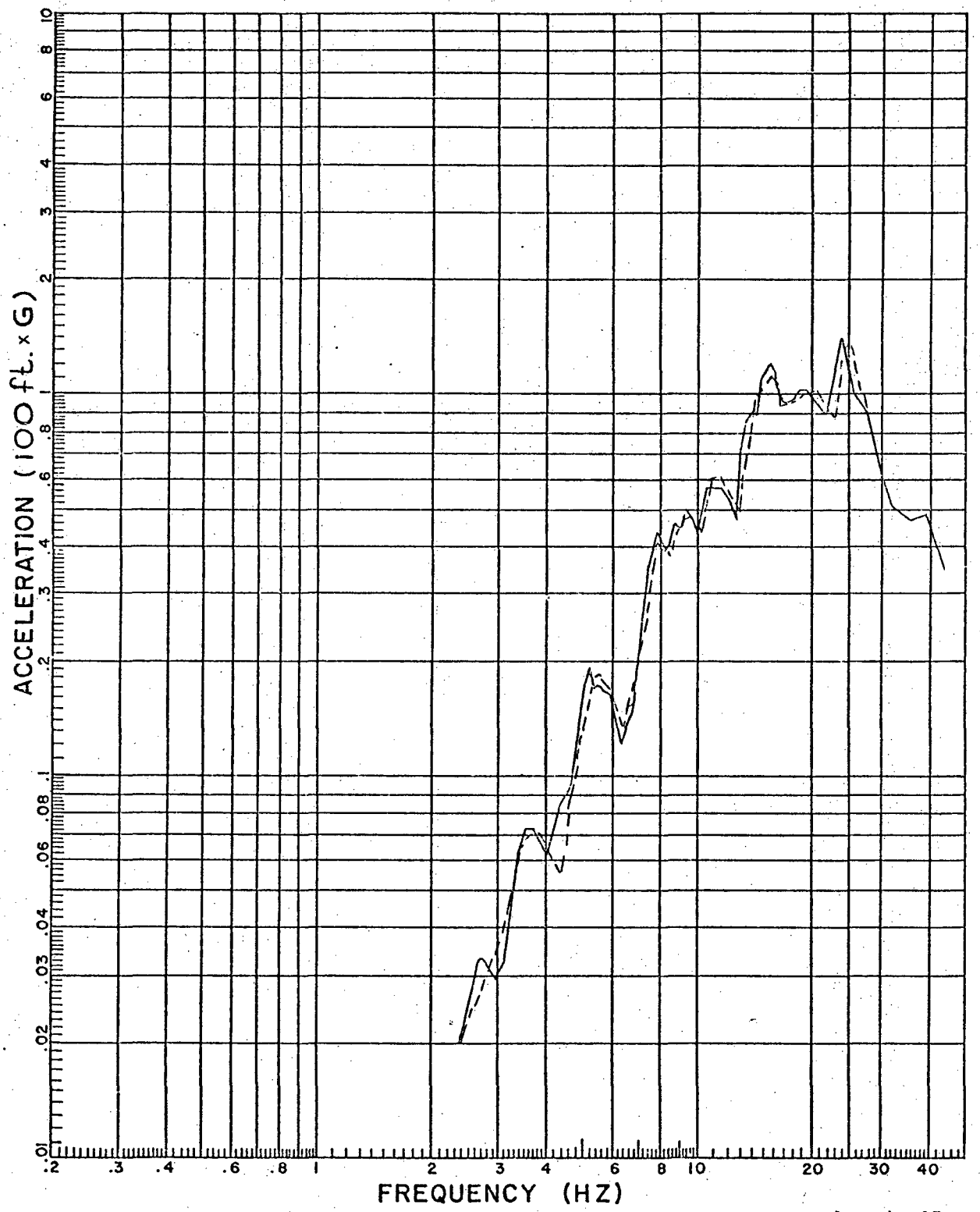

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FIGURE A.49-4

HETHA - ANSYS COMPARISON

NODE 1 - TORSION

ACCELERATION RESPONSE SPECTRA 3% DAMPING

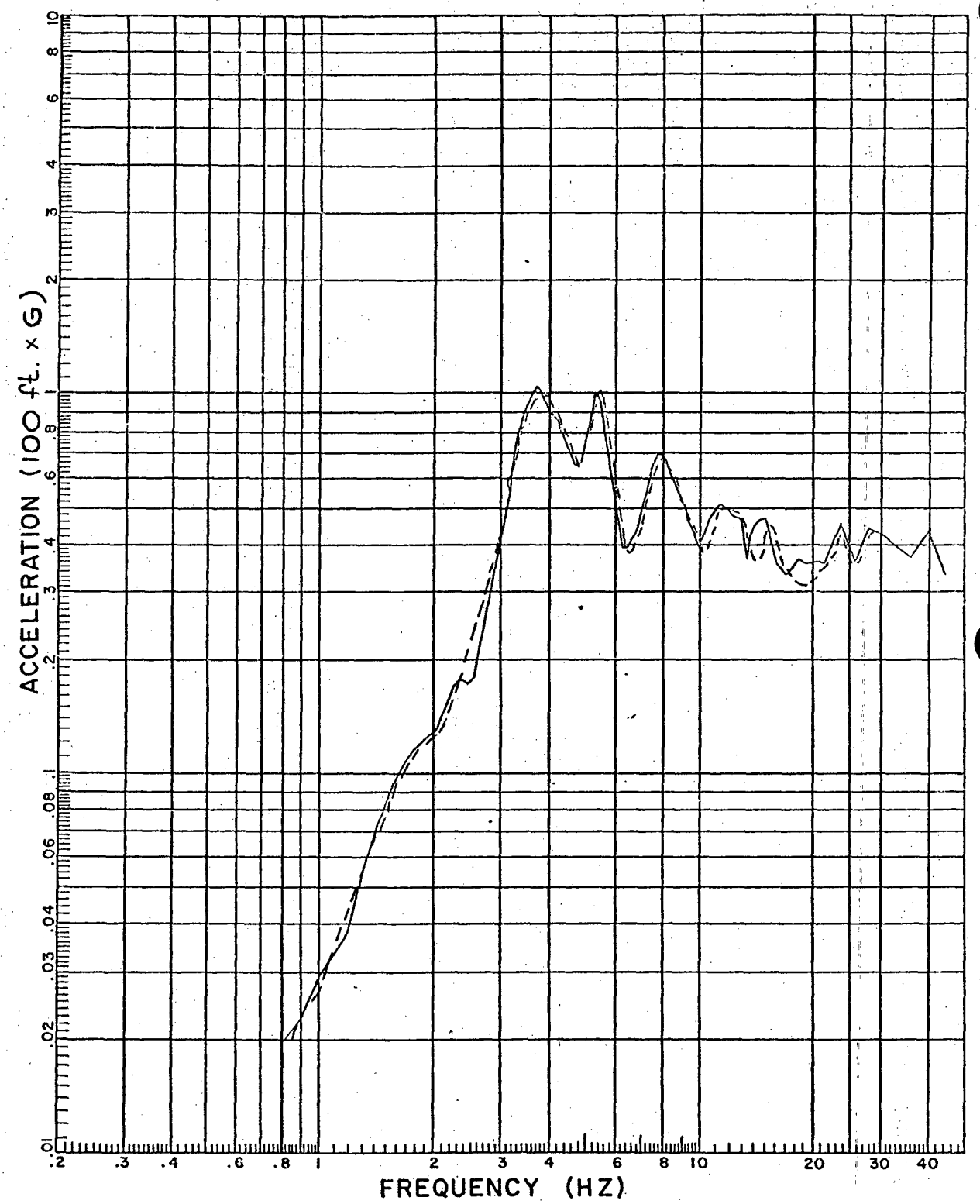

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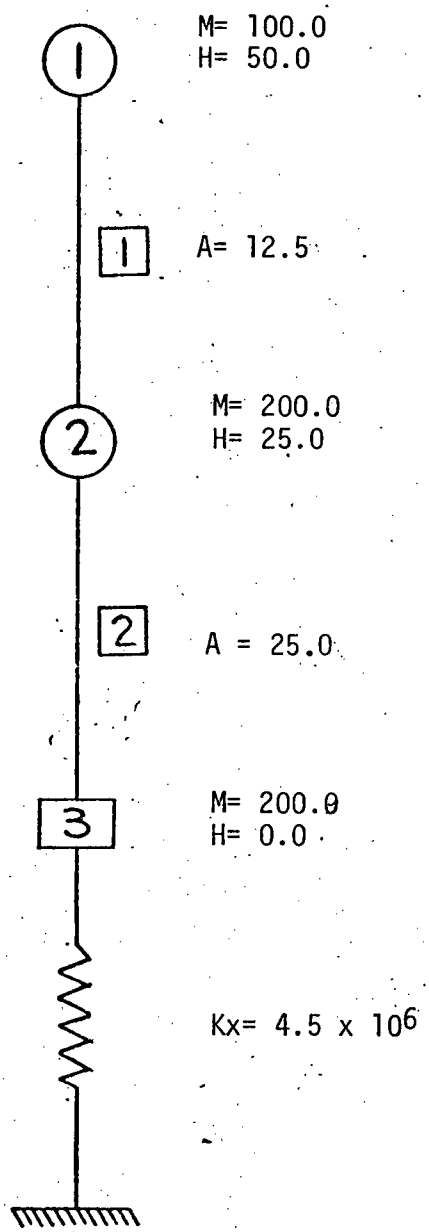
FIGURE A.49-5

VETHA - ANSYS-MATHEMATICAL MODEL



$E=5 \times 10^5$ K/ft^2

Notes:

1- For damping values see Figure 7 A.49-7
2- For symbols and units see Table III A.49-3

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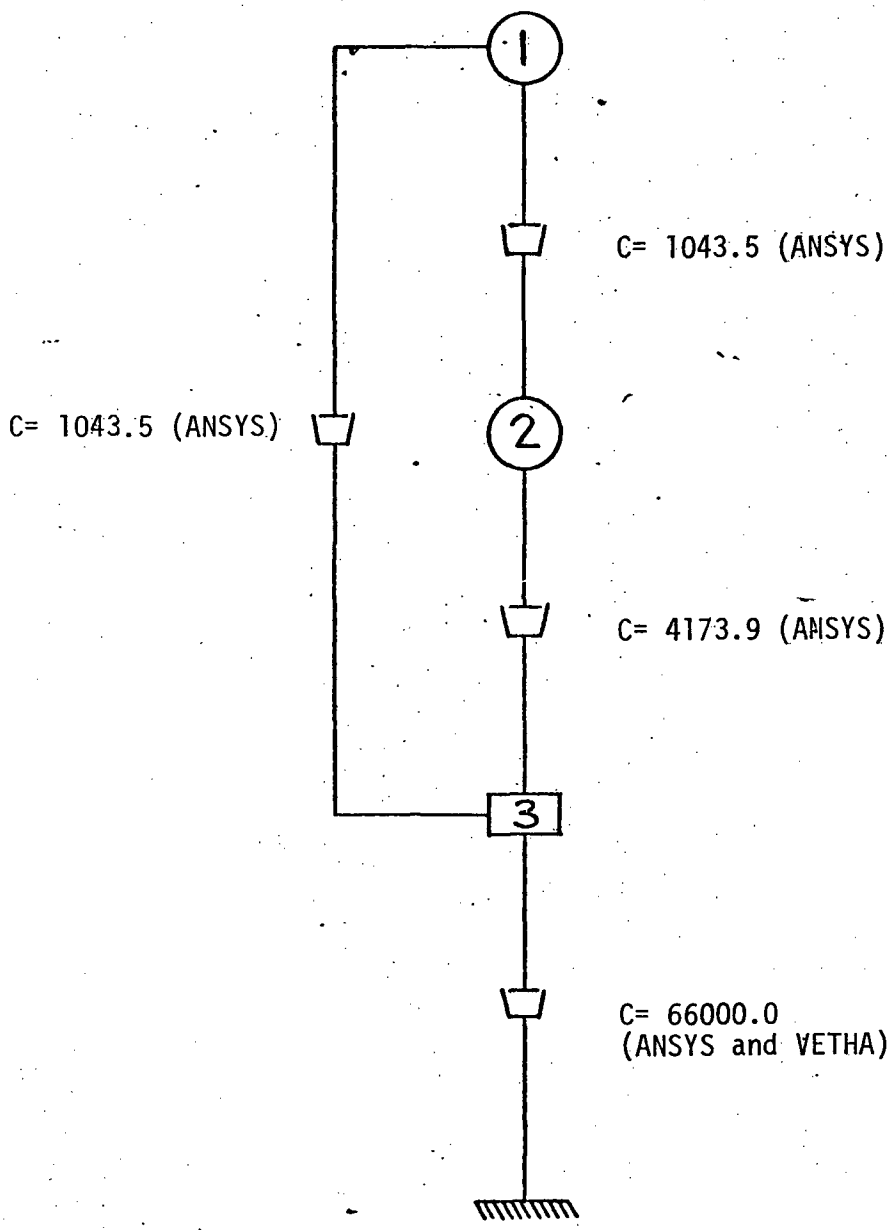
FIGURE A.49-6

VETHA - ANSYS-MATHEMATICAL MODEL-DAMPING VALUES



Notes:

1- Structural damping for VETHA: 7% of critical based on a "fixed" base condition
2- For stiffness and mass properties, see Figure A.49-6
3- For symbols and units, see Table A.49-3

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FIGURE A.49-7

VETHA - ANSYS COMPARISON

NODE 1 - VERTICAL TRANSLATION

ACCELERATION RESPONSE SPECTRA - 3% DAMPING

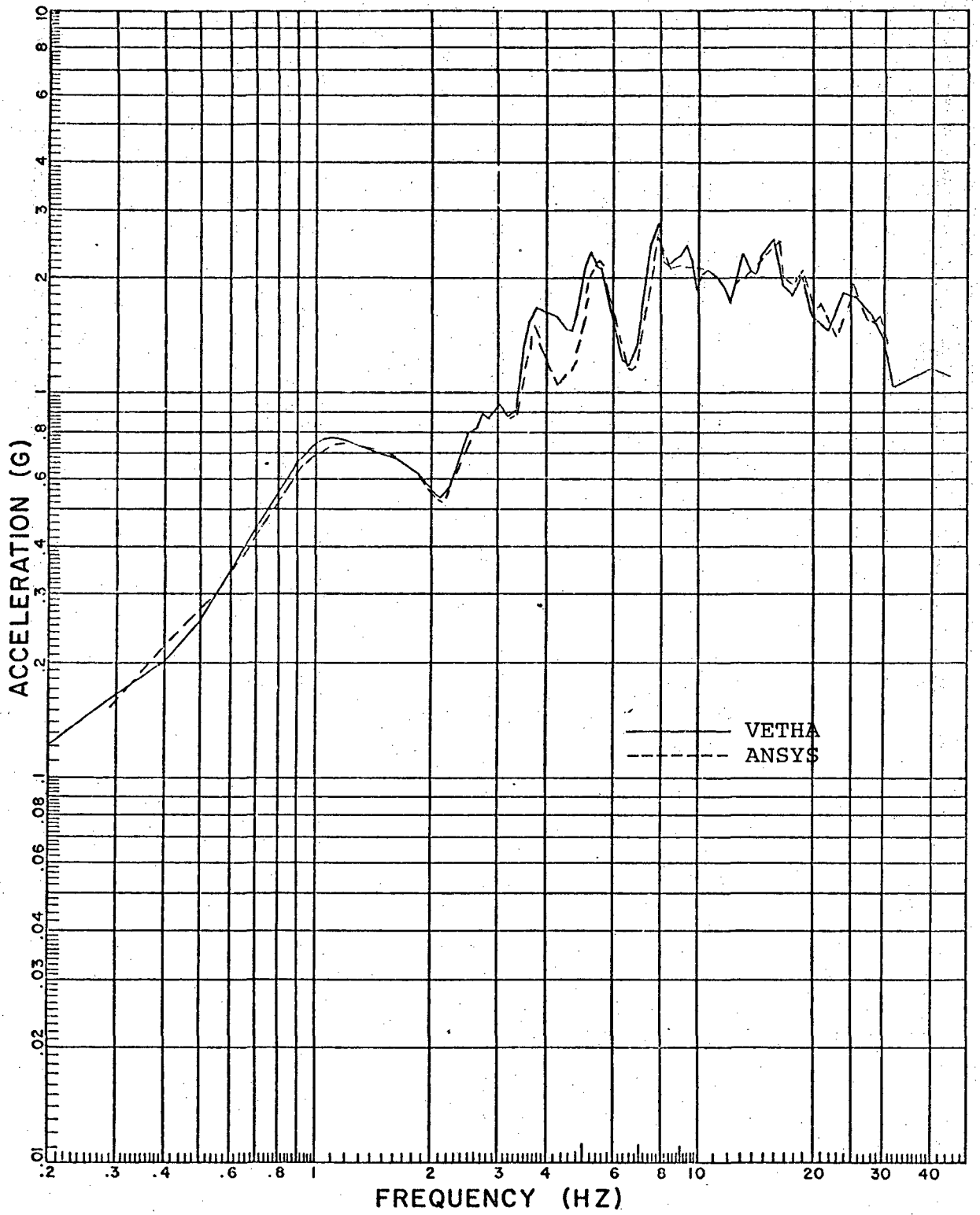

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FIGURE A.49-8

Application

HETHA and VETHA are used for the seismic analysis of the CRBRP structures. HETHA for horizontal input motions, VETHA for vertical.

Reference

Tsai N. C., "Modal Damping for Soil-Structure Interaction" ASCE, Journal of the Engineering Mechanics Division, April 1974, Pages 323 through 339.

A.50 HOTDAMG

The HOTDAMG program may be used to perform a variety of calculations related to transient thermal stress evaluation. It provides a rapid calculation of the creep and fatigue damage of components based on the simplified inelastic approach described in references below. The normal method of calculation determines the transient response of a flat plate subjected to typical reactor transients for a series of events which combine to form a duty cycle. It can also be used for rapid scoping calculations of thermal transients only or stress calculation only to determine the damage of specified initial residual stress, maximum strain and operating temperature.

Availability

HOTDAMG has been available on the CDC 7600 Computer at Westinghouse Monroeville Center (MNC) division.

Verification

Partial verification by comparison with other codes is contained in the references cited below. Additional verification will consist of comparison with other recognized and accepted codes such as ANSYS.

Application

Damage calculations for upper internals structure components, such as: upper and lower plates, columns, shroud tubes, instrumentation posts and thermal shieldings. Also used for core barrel, horizontal baffle, control rod lines and other high temperature components.

Reference

1) WARD-D-0099, "Simplified Cumulative Damage Evaluation of CRBRP Thermal Transients", May 1975.

2) WARD-D-0048, "A Simplified Cumulative Damage Evaluation of Fast Breeder Thermal Transients to the Criteria of ASME Code Case 1331-8", May 1975.

A.51 HYTRAN

HYTRAN is a computer program to predict hydraulic transient phenomenon for complex piping systems subjected to external forcing functions emanating from blasts and seismic disturbances. HYTRAN can calculate hydraulic transient pressure in circuits and determine the peak magnitudes of hydraulic transient pressure. It can also identify locations within the circuit where the peak pressure and/or column separations occur, plus size and locate attenuators to protect equipment and other circuit elements. HYTRAN was developed by Ralph M. Parsons Company for U.S. Army Corps of Engineers, Huntsville Division.

Availability

HYTRAN has been available on the CDC 7600 computer of Lawrence Berkeley Laboratory. The version was released by Waterways Experiment Station, Corps of Engineers, Department of the Army, Vicksburg, Mississippi, June, 1973.

Verification

When HYTRAN was being developed, an experiment test program was initiated by USAEDH in 1969 to supplement the analytical study. The test result verified the validity of HYTRAN. The documentation, including description of the sample problems, is in Reference 1. A comparison of HYTRAN results with experimental results, is shown in Figure A.51-1.

Application

HYTRAN can be used in a complex piping system for determining the pressure transient in the circuit due to sudden disturbance. It is currently used in the study on pressure transients on the rupture discs due to seismic events.

Reference

"Hydraulic Transient (HYTRAN) Computer Program Users Manual", HNDTR-73-9-ER-R, U.S. Army Corps of Engineers, Huntsville Division, May, 1973.

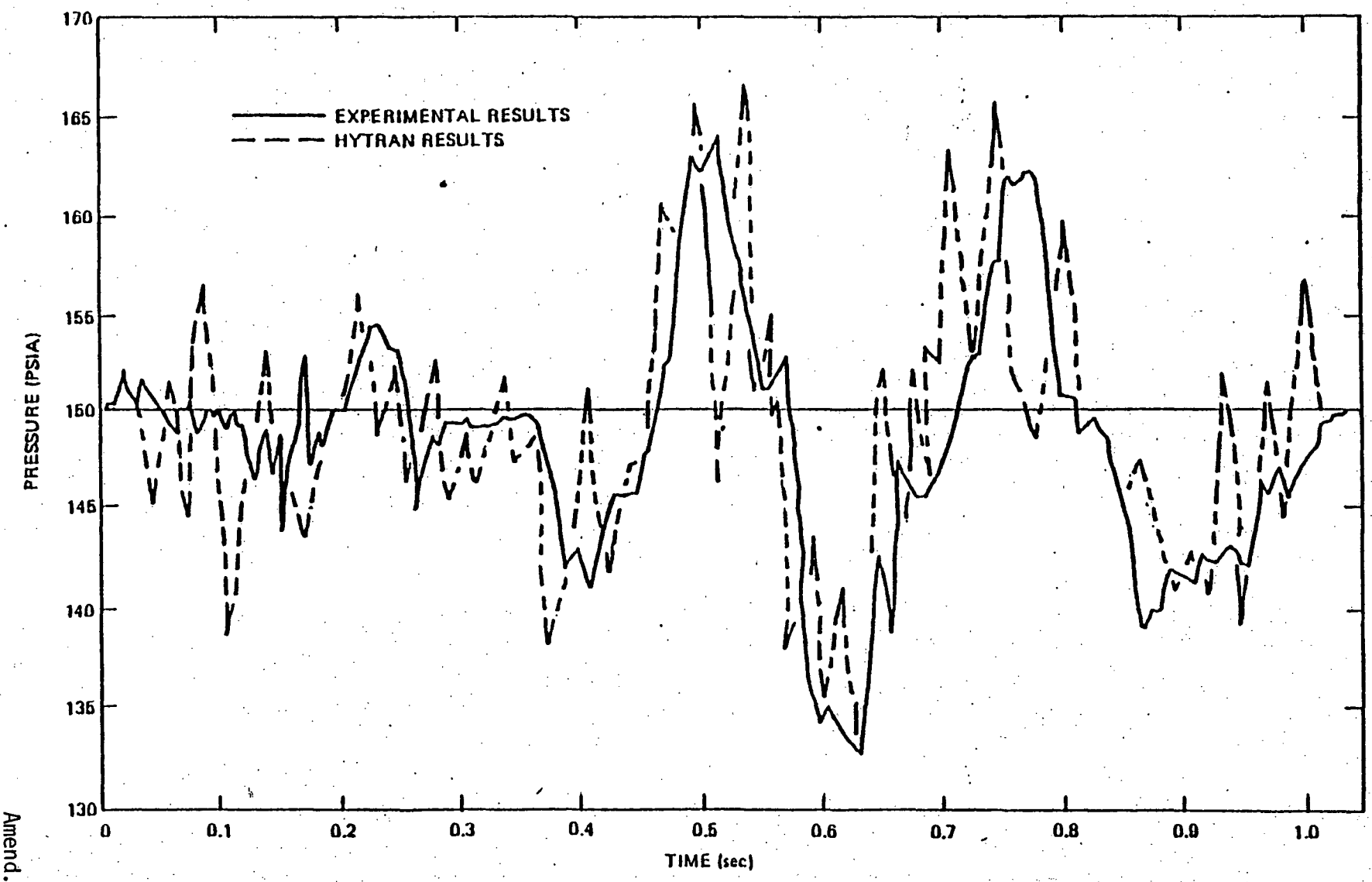

Figure A.51-1 *Comparison of HYTRAN and Experimental Results*

A.52 KALNINS

The program calculates the stresses and displacements in thin-walled elastic shells of revolution, when subjected to static edge, surface and/or temperature loads with arbitrary distribution over the surface of the shell. The geometry of the shell must be symmetric but the shape of the meridian is arbitrary. It is possible to include up to three branches, if these branches are also symmetric. In addition, the shell wall may consist of four layers of different orthotropic materials and the thickness of each layer and the elastic properties of these layers may vary along the meridian. Since the program is based on classical shell theory, it has the same limitations, namely:

1. The material is linearly elastic.
2. All displacements are very small.
3. Both the stress and strain normal to the surface of the shell are so small that they may be neglected entirely.
4. The shearing strains through the thickness are negligible so that normals to the midsurface remain straight and normal after deformation.
5. Both principal radii are much greater than the shell thickness. Generally $\frac{R}{t}$ must be greater than 10 for adequate accuracy.

Availability

The February 1976 version of KALNINS is available on the IBM 370 Model 165 computer of the Chicago Bridge and Iron Company.

Verification

Verification of this program is achieved through the use of the following problems.

1. Comparison of 2/1 ellipsoidal and torispherical heads.

   The problem consists of comparing a 2/1 ellipsodial head to an equivalent torispherical head subjected to the same uniformly distributed internal pressure. An equivalent torisphere is defined as one having the same height above the tangent line as the ellipsoid and a minimal L/b ratio. See Figure A.52-1 for geometry of torishperical and ellipsoidal heads.

   Plots of the hoop force and longitudinal bending from KALNINS' results compare the ellipsoidal and torispherical heads. Even though the change in radii has been minimized, the disturbance at the junction of the sphere and torus is considerable (see Figure A.52-2). See Figures A.52-3 thru A.52-7 for plots of Ø and θ on the inside, outside and meridian of the head.

2. Cylindrical Water Tank with Tapered Walls

The problem used for this verification is "Shell of Variable Thickness" taken from "Stresses in Shells" by W. Flugge, pages 289-294, Springer-Verlog, New York 1973.
The problem consists of a tapered shell filled with water. The shell has a radius of 9' 0" and is 12' 0" high. The shell thickness varies from 11" at the bottom to 3" at the top. See Figure A.52-8 for the Z axis.

KALNINS results compare favorably with the theoretical solution as noted below:

KALNIN'S	Theoretical Solution "Stresses in Shells"
346.8 lb/in 4160 lb/ft	4180 lb/ft

KALNIN'S	Theoretical Solution "Stresses in Shells"
-1539 in-lb/in=-1539 ft-lb/ft	-1470 ft-lb/ft

Figure A.52-9 shows the location of the Ø and θ axes and Table A.52-1 gives a comparison of Final Results for Nθ and MØ.

3. Circular Hole in Plate

The problem used for this verification is presented in"The Effect of Circular Holes on Stress Distribution in Plates" taken from "Theory of Elasticity" by Timoshenko & Goodier pages 90-97 McGraw Hill, New York, 1970. The problem consists of a plate with a circular hole, submitted to a uniform tension in the direction of the "X" axis, the results show that the prpgram gives a 0.02% deviation from the theoretical solution.

Kalnin's	Theoretical Solution "Theory of Elasticity"
Nθ = 3.092	Nθ = 3.0915

4. Inclined Cylinder under Hydrostatic Loading

The problem used for this verification is "Inclined Cylinder" taken from "Stress in Shells" by W. Flugge pages 114-118, Springer-Verlag New York,1973. The problem consists of an inclined cylinder partially filled with water. The uniform cylinder has a radius of 100", length of 300" and a thickness of 5/16 (see Figure A.52-10).

Table A.52-2 compares the final results.

Application

KALNINS will be used in the design of the closure head assembly to check stresses in the support skirts.

TABLE A.52-1

Program KALNINS Results			"Stresses In Shells" Solution At Maximums
Distance From Base	Nθlb/in.	Mϕ $\frac{\text{in.-lb}}{\text{in.}}$	
0.0	5.919×10^{-6}	→ -1539.0	Mϕ=-1470ft-lb/ft
6.0	21.15	-903.9	
12.0	71.29	-440.5	
18.0	134.0	-124.8	
24.0	194.3	71.47	
30.0	253.3	177.1	
36.0	297.2	218.3	
42.0	327.3	217.6	
48.0	343.3	192.8	
54.0	→346.8	157.1	Nθ=4160 lb/ft
60.0	339.6	119.5	
66.0	324.2	85.46	
72.0	303.0	57.80	
78.0	277.9	36.29	
84.0	250.8	23.41	
90.0	222.9	15.00	
96.0	195.1	10.58	
102.0	167.8	8.685	
108.0	141.4	8.075	
114.0	115.9	7.754	
120.0	91.45	7.032	
126.0	68.13	5.584	
132.0	46.29	3.453	
138.0	26.50	1.177	
144.0	94.53	-1.481×10^{-3}	

Comparison of Final Results for Nθ and Mϕ

TABLE A.52-2

	$\theta = 180^{o}$, $\xi = -1$ (S = 50")			$\theta = 180^{o}$, $\xi = 0$ (S = 150")		
	Theoretical Solution	Program KALNINS	% Deviation	Theoretical Solution	Program KALNINS	% Deviati
$N\phi$	-523.8	-513.7	1.92	-130.9	-125.9	3.8
$N\theta$	523.8	523.7	0.01	261.89	261.4	0.19
	$\theta = 240^{o}$, $\xi = -1$ (S = 50")					
$N\phi\theta$	340.2	340.3	0.03			
	$\theta = 225^{o}$, $\xi = 0$ (S = 50")					
$N\phi\theta$	130.9	129.7	0.92			

Final Compared Results

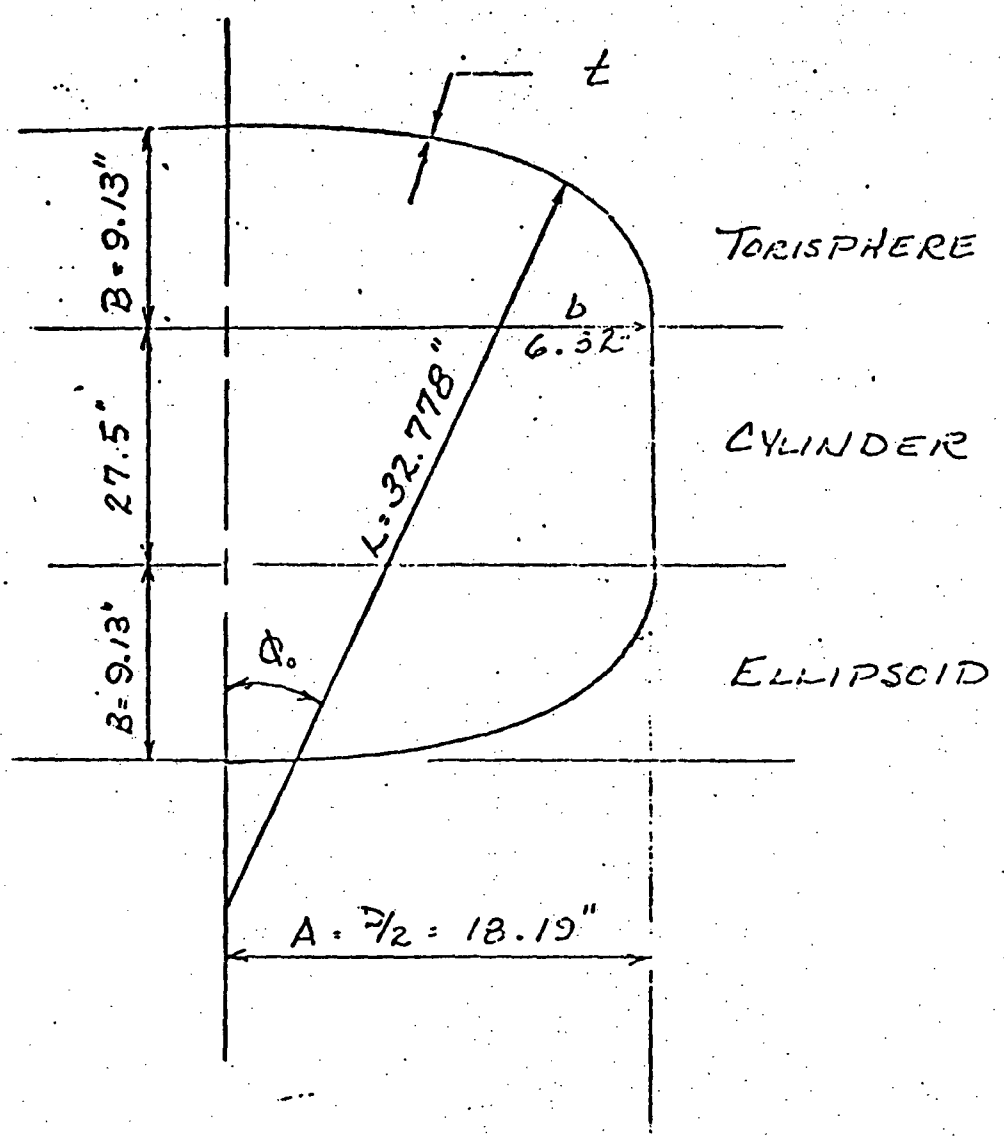
t
B = 9.13"
TORISPHERE
b
6.32"
L = 32.778"
27.5"
CYLINDER
B = 9.13"
ϕ_0
ELLIPSOID
A = D/2 = 18.19"

Figure A.52-1

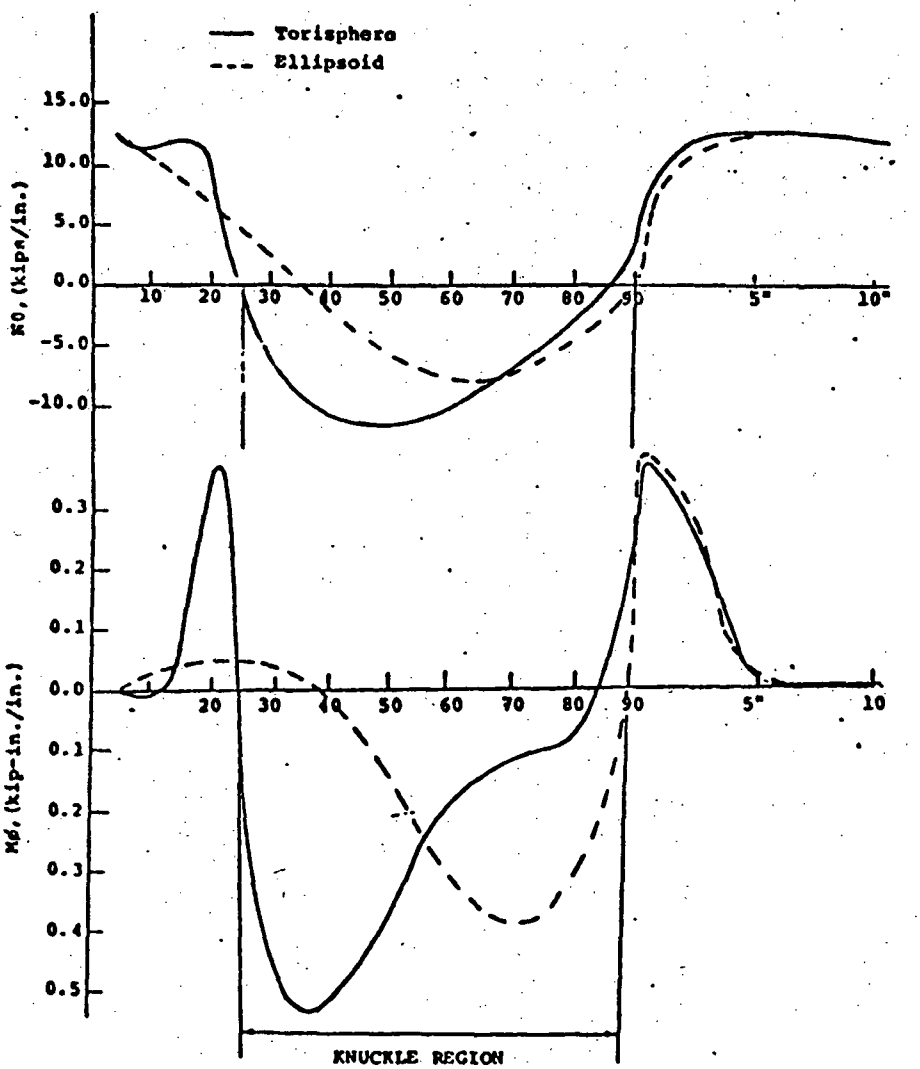

Fig. A.52-2 Plot Of Nθ And Mφ From KALNINS Output

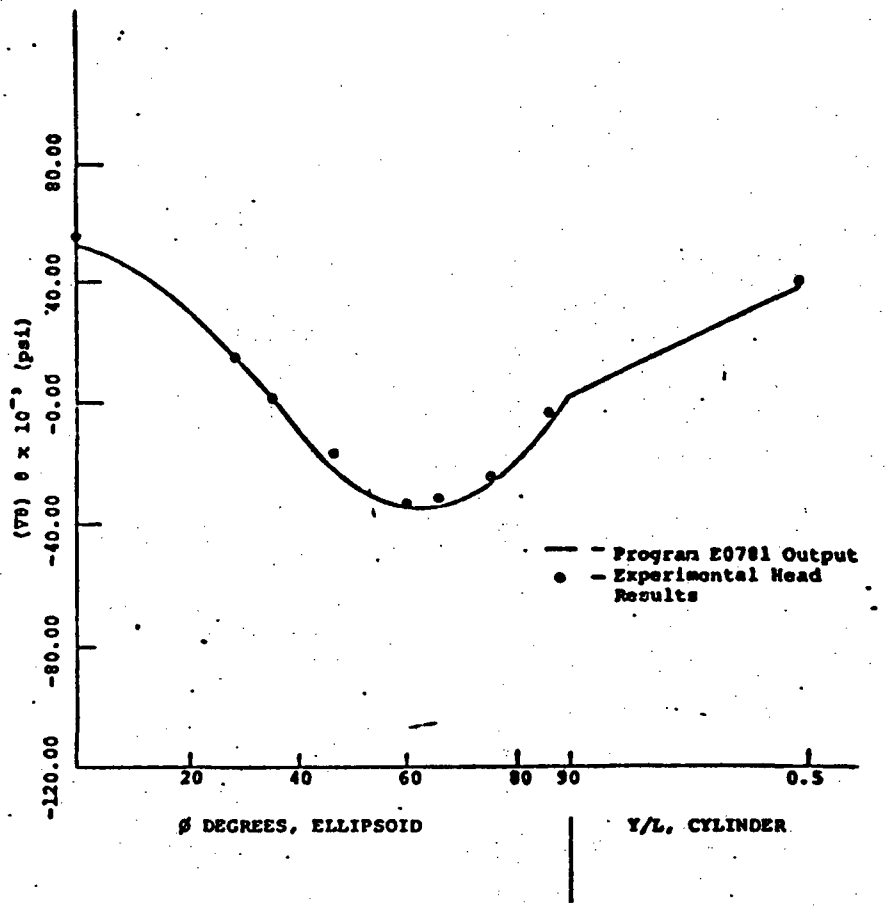

Figure A.52-3

Plot of Stress In The θ Direction On The Outside Surface (θ = 0)

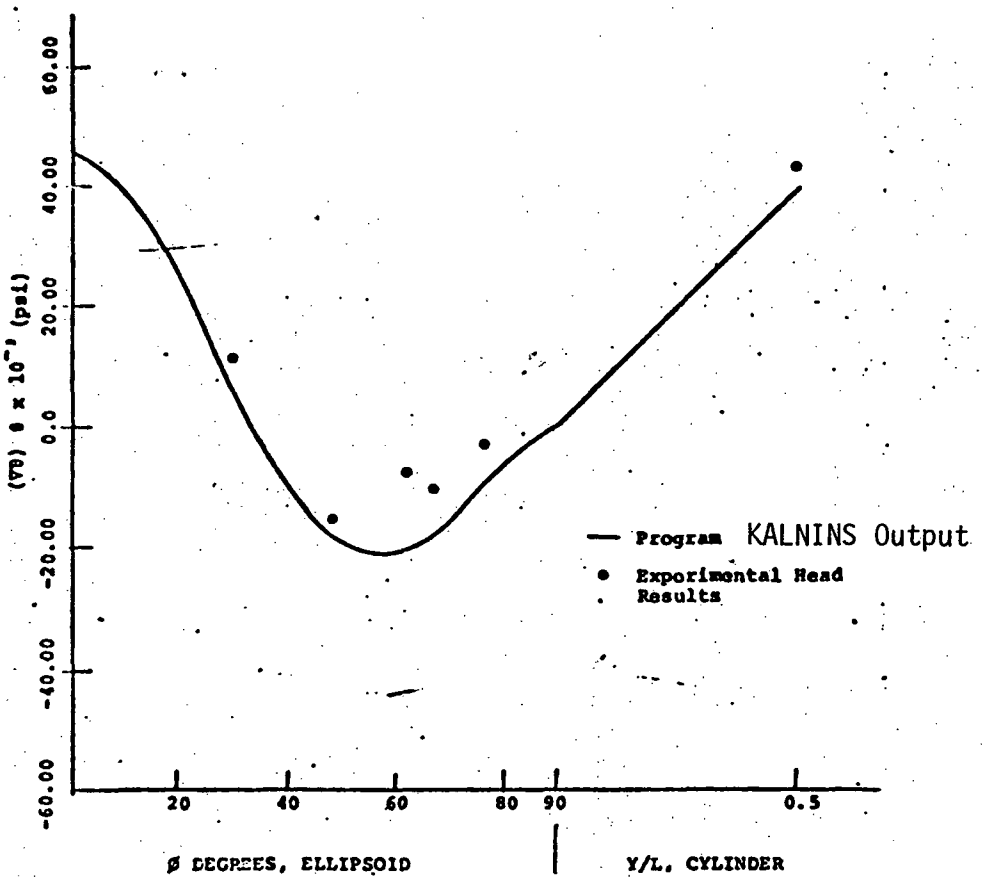

Fig. A.52-4
Plot Of Stress In The θ Direction
On The Inside Surface (θ = 0°)

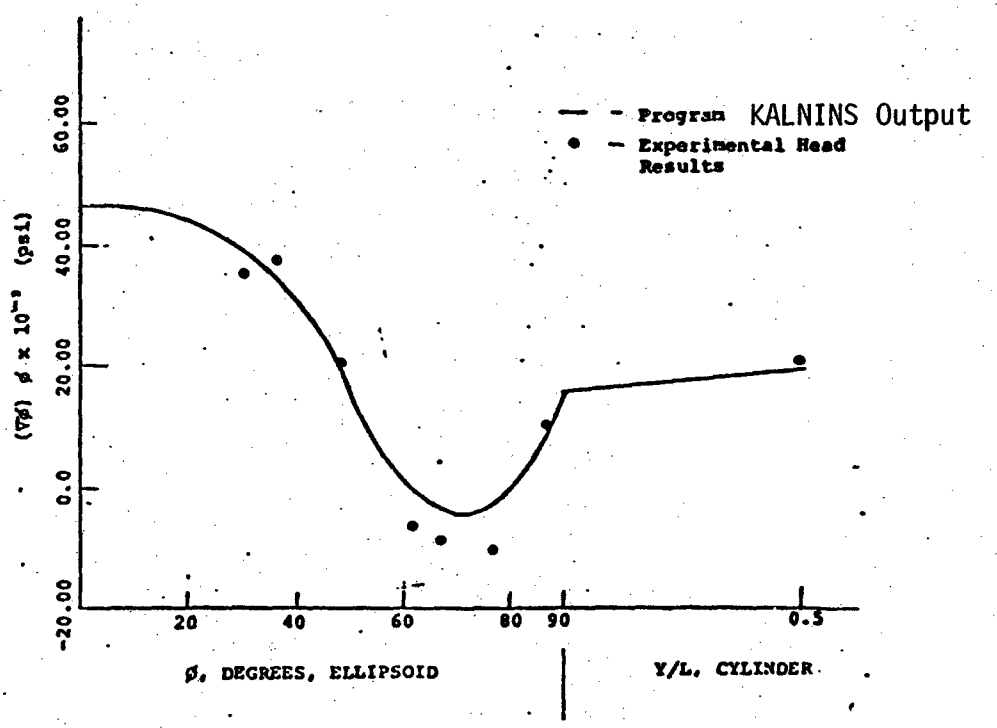

Fig. A.52-5

Plot of Stress In The ø Direction On The Outside Surface (θ = 0°)

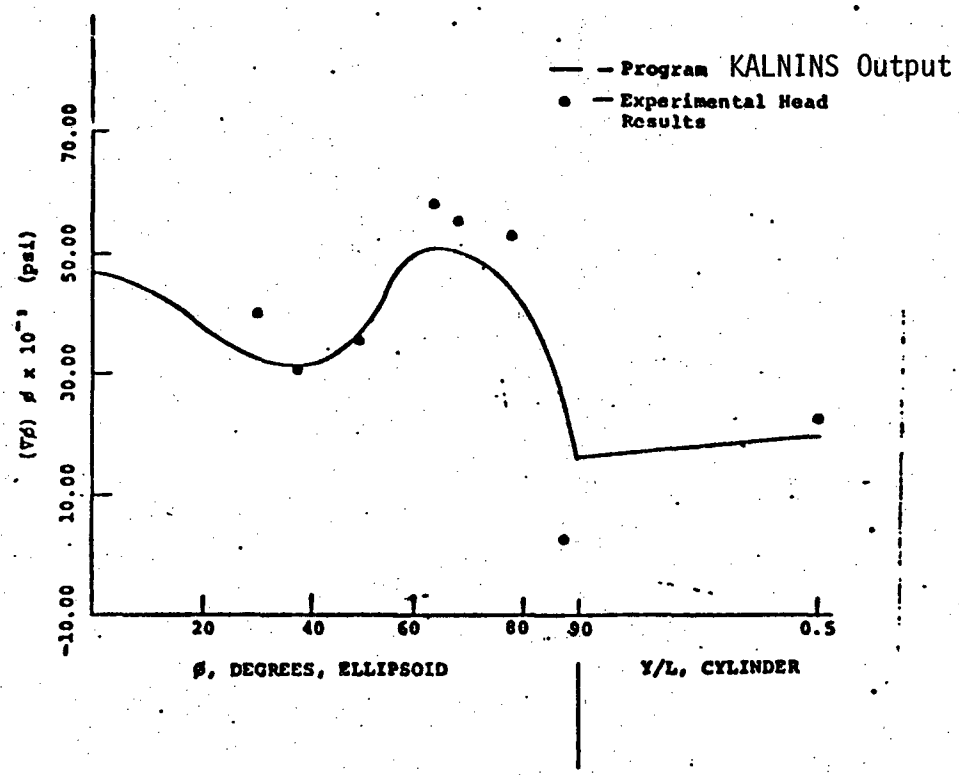

Fig. A.52-6
Plot of Stress In The φ Direction
On the Inside Surface (θ = 0°)

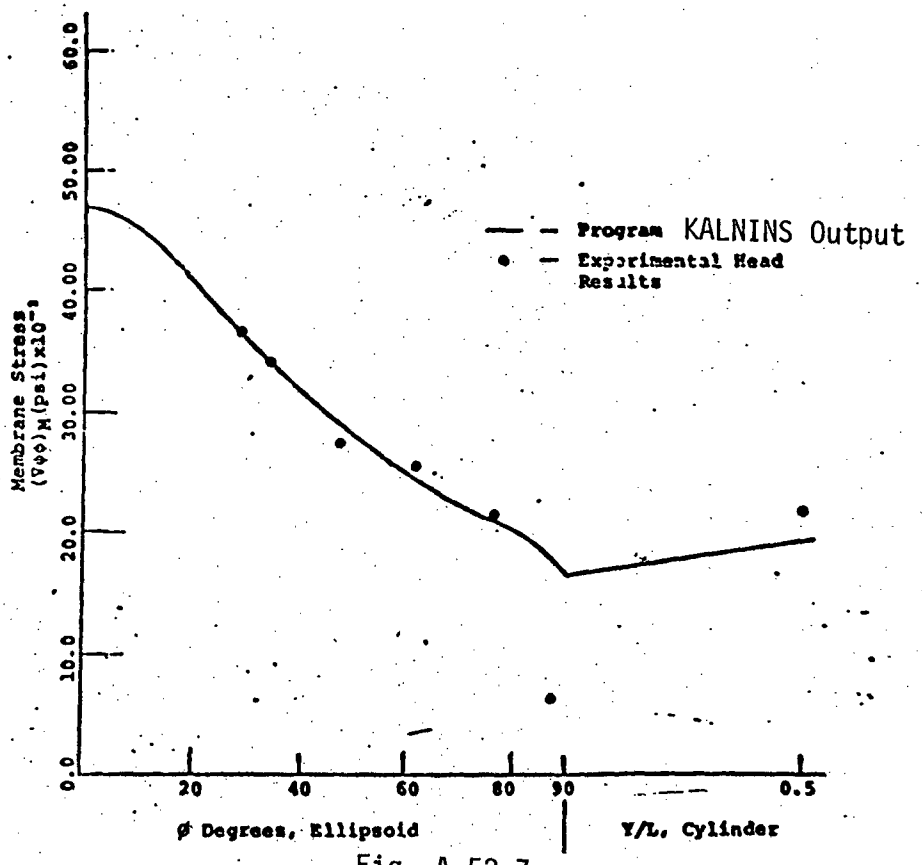

Fig. A.52-7

Plot Of Membrane Stress ($\theta = 0^{\circ}$)

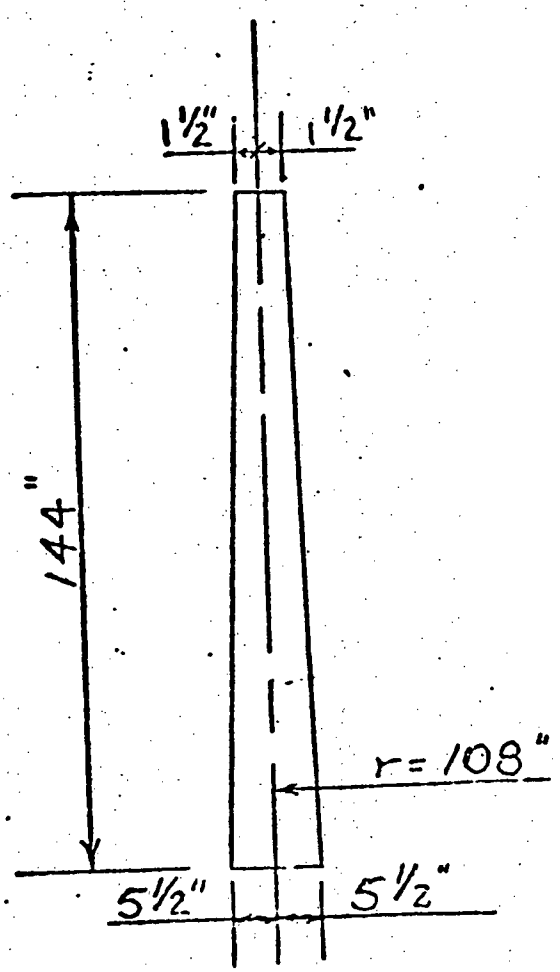

Figure A.52-8

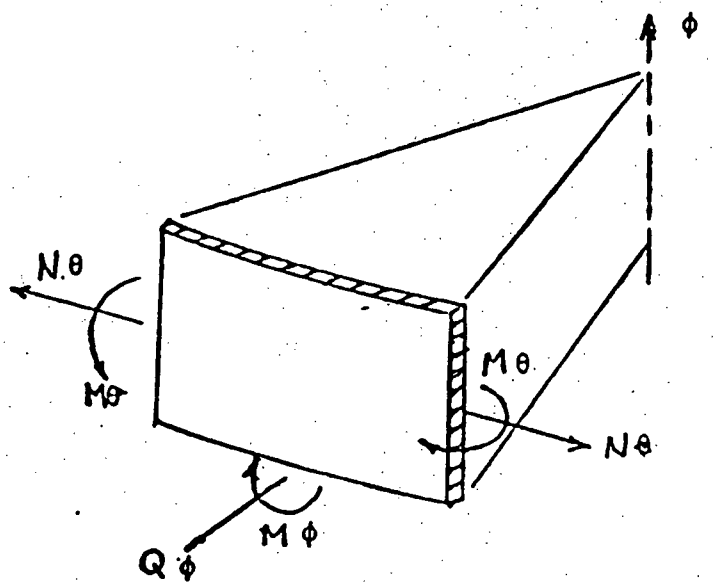

FIG. A.52-9
Location of ϕ and θ axis

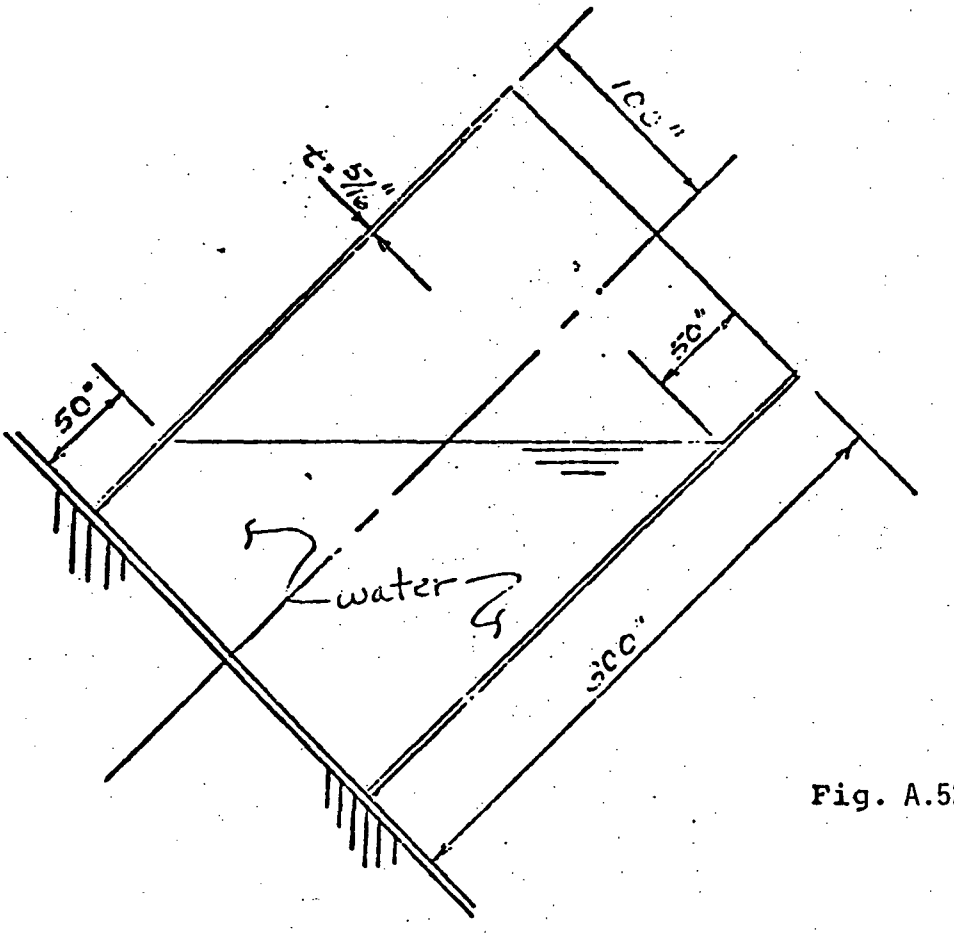

Fig. A.52-10

A.53 KENO-IV

KENO-IV is a multigroup Monte Carlo program to predict the criticality of arrays of fissionable material. The geometry options permit descriptions of complex three-dimensional systems. The results include the mulitplication factor, region-dependent fission densities, and energy-and region-dependent fluxes.

Availability

KENO-IV is available on the IBM 360 system of Rockwell International. It has been employed by the Atomics International Division and Rocky Flats plant.

Verification

KENO has been validated by comparison to 35 high-enriched critical units and arrays(1) and 40 low-enriched uranium critical systems by the Oak Ridge Y-12 plant. A verification against the ZPPR-2 critical assembly was made by the AI Rocky Flats plant. A description of the verification methods and procedures used is presented in AI Report N099TI414028, "Evaluation of the Upper Limit Criticality of the Ex-Vessel Storage Tank (EVST) for the CRBRP," J. Otter, January 1978.

Application

KENO-IV is currently used for determining the criticality of possible ex-vessel storage tank loadings,

Reference

"KENO-IV, An Improved Monte Carlo Criticality Program," by L. M. Petrie and N. F. Cross, ORNL-4938, Oak Ridge National Laboratory, Oak Ridge, Tennessee, November 1975.

A.54 LIFE-III

The LIFE-III code (Ref. 1) is a fuel pin behavior analysis code which was originally developed at the Argonne National Laboratory. The code has been selected as the national fuel pin modeling code and is undergoing further development at ANL, W-ARD, HEDL and GE. The code calculates the thermal state of the fuel and mechanical behavior of the fuel rod as a function of operating history. The stresses at the fuel-cladding interface and the strains in the cladding are analyzed as a function of burnup, power, cladding temperature and fuel parameters such as density and fuel-cladding gap size. Cladding loading results from fission gas pressure and fuel-cladding mechanical interaction. The cladding strains are separated into components of swelling, irradiation creep and thermal creep. The radial temperature profile across the fuel and the restructuring of the fuel using a pore migration model are also analyzed as a function of burnup, power, loading temperature and fuel parameters. The various fuel-cladding behavior models in the code are based on available data and the overall code is calibrated using experimental fuel pin results.

Availability

The current version of the code is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

The LIFE-III code has been successfully calibrated, supporting documentation has been prepared, program checkout is completed and a released version was transmitted to the Argonne National Code Center. The LIFE-III code was utilized for CRBRP fuel rod performance analysis. As irradiation data and/or improved fuel and cladding models become available, the LIFE-III code will be periodically updated. The latest released and documented version will be used for CRBRP fuel rod analysis in support of the FSAR.

Application

The LIFE-III code has been used to calculate fuel and blanket rod behavior, particularly to verify compliance with the no-incipient melting design criterion and to calculate fuel-cladding contact loads during steady state operation.

References

53 1. ERDA-77-56, "LIFE-III Fuel Element Performance Code", ANL, July 1977.

A.55 LION

LION is a digital computer program which will solve three-dimensional transient and steady-state temperature distribution problems. The input consists of gemetry, physical properites, boundary conditions, internal heat generation rates, and coolant flow rates as a function of time. In addition to solving problems of heat conduction in a structure, LION can handle forced convention, free convection, and radiation or a combination of these at the surface of the structure. The output consists of complete nodal temperature distributions along with surface fluence and surface heat transfer coefficients. An option is included in the program for determining the mean temperature in any specified section of the structure.

Availability

LION is available from the Argonne National Laboratory Code Center and is used on the CRBRP project by the Foster Wheeler Energy Corporation.

Verification

Results from the LION code have been checked against analytic solutions of typical problems and compared with results from the CINCA-3G Code (A.10).

Application

LION is used to determine temperature distributions within the PHTS check valve.

Reference

Schmidt, J. R., Lechliter, G. L., Fischer, W. W., "LION: Temperature Distributions for Arbitrary Shapes and Complicated Boundary Conditions," KAPL-M-6532, Knolls Atomic Power Lab., Schenectady, N. Y., July 27, 1966.

A.56 LSD-2 (Proprietary Burns and Roe)

Line Source Dose - 2 calculates the dose rate behind a multilegged shield due to a number of gamma emitting cylinderical sources. LSD - 2 employs the Rockwell method of approximating the dose rate from a cylindrical source behind a semi-infinite slab shield. The cylindrical source is replaced by an equivalent line source which is located inside of the source to account for source self-attenuation.

Given the radiation sources and the shield material thicknesses along the line of sight from the radiation sources to the detector point, the corresponding gamma exponential attenuation is determined for each source energy group to yield the uncollided gamma flux contribution. The collided flux contribution is calculated by the use of a single material buildup factor. A Lagrangian interpolation of gamma flux to dose rate conversion factors is used to yield the dose rate.

Availability

LSD-2 is a Burns and Roe proprietary code.

Verification

The LSD-2 code verification has been by hand calculation, as recorded in Burns and Roe, Inc. proprietory documents.

Application

LSD-2 is used to determine bulk shielding wall thicknesses to meet the CRBRP radiation zone requirements.

A.56.A NUPIPE

NUPIPE is a linear elastic, static, and dynamic finite element computer code for three-dimensional piping systems. The code is primarily used to perform thermal expansion, deadweight, and seismic analysis of the Auxiliary Liquid Metal Piping System. The program performs analysis in accordance with the requirements of the ASME Section III Code for Nuclear Class 1, 2, or 3 components and ANSI B31.1 for power piping at the option of the user.

Availability

NUPIPE Version 1.4 is currently available on the IBM 370/3033 computer of Rockwell International, Canoga Park, California.

Verification

Since 1976, NUPIPE has been used for piping analyses of over 50 licensed U.S. nuclear power plants. The NRC example problems have been run and results approved by NRC. Additional hand calculations and cross-checks with other piping analysis programs have further verified the correctness of the program.

Application

NUPIPE will be utilized for the static and dynamic analyses of piping systems of the Auxiliary Liquid Metal System.

Reference

Quadrex Corporation, 1700 Dell Avenue, Campbell, California 95008, "Verification Manual for NUPIPE Version 1.4," "User's Manual for NUPIPE Version 1.4," and "Programmer's Manual for Version 1.4."

A.57 MAP

The MAP code is a radiation transport code employing point kernel techniques with a multigroup angular dependent surface source. The surface source geometry is the cylindrical surface defined by DOTIIIW (see A.22) discrete ordinate transport code problem. Angular-and energy-dependent surface source data are obtained from the DOTIIIW code on magnetic tape and processed by the MAP code to provide flux and response data at a surface detector. MAP provides techniques which circumvent the use of discrete ordinate transport codes in calculating radiation transport through voids or near-voids.

Availability

The MAP code is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center. The version currently being used was released from WANL in August, 1970, and has been updated at ARD to satisfy CRBRP shield design analysis needs.

Verification

The MAP code results have been compared to results from hand calculations, results from similar codes, and experimental results. The use of MAP to accurately predict the detector response at various positions external to experimental configurations demonstrate the validity of the MAP technique.

Application

MAP is used to predict neutron spectra and radiation detector responses behind experimental configurations which have been modeled in DOTIIIW discrete ordinates transport claculations. MAP results are then compared to experimental results in order to verify or to define problem areas in CRBRP shielding design analysis methods and/or basic nuclear data.

Reference

R. G. Soltesz, R. K. Disney, J. Jedruch, and S. L. Zeigler, "Nuclear Rocket Shielding Methods, Modification, Updating, and Input Data Preparation. Volume 5. Two-Dimensional, Discrete Ordinates Transport Technique. Final Progress Report." WANL-PR(LL)-34, Vol. 5 (NASA-CR-102968), August 1970.

A.58 MARC

This is a general purpose, three-dimensional, non-linear finite element computer program. Like the ANSYS (A.3) program, it is capable of static and dynamic, elastic and plastic, creep and heat conduction analyses. The program has large libraries of elements and an extensive selection of material behaviors, both linear and nonlinear, so that the software serves a wide spectrum of uses, from linear elastic analysis of 2- and 3-dimensional solids, shells and beams to applications in which non-linear material and geometric effects dominate and must be included in conjunction with sophisticated geometric modelling.

Availability

The production versions of the program most recently released by MARC Research Analysis Inc. are available through the CDC 6600 computer of Cybernet Center at Palo Alto, California. GE-FBRD will use the computer facilities of Lawrence Berkeley Laboratory after the programs are installed on their CDC 7600 in early 1976. The version of MARC released in March 1975 is used on the Westinghouse Power Systems CDC-7600 at the Monroeville Nuclear Center.

Verification

MARC is recognized and widely used in industry with a sufficient history of successful applications to justify its validity per SRP 3.9.1, Section II.2.a.

Application

It will be extensively utilized in studying the elastic and inelastic behaviors for the steam generators, sodium pumps, transient joints, reactor components, piping, etc., to ensure their structural integrity.

References

1. "MARC-CDC User Information Manual (Volume I)," Publication No. 17309500, Control Data Corporation, Minneapolis, Minnesota, June, 1974.

2. "MARC-CDC Demonstration Problems Manual (Volume III)," Publication No. 17311700, Control Data Corporation, Minneapolis, Minnesota, October, 1974.

A.59 MENAC (Westinghouse Proprietary)

MENAC code calculates the steady state performance of the nuclear steam supply (NSS). The code is a single loop model of the thermal hydraulic characteristics of the NSSS including the pump characteristic, system pressure drops, heat exchangers (IHX's and steam generator modules) and the steam generator recirculation loop. The code calculates nominal, pessimistic, and optomistic plant conditions and has the capability to perform a statistical evaluation of plant operation based on randomly selected values of performance related parameters within their uncertainty conditions due to variations on specific performance parameters, and to quantify the margins between design and expected operating conditions.

Availability

MENAC is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

The MENAC code will be verified by comparing IHX ans SHS thermal hydraulics calculations with steady state vendor furnished results, and hand calculations of system pressure loss/pumping capability interactions. Final validation of results will depend upon the prototype tests for the pumps and steam generator modules.

Application

MENAC was used to calculate different sets of plant conditions corresponding to various component thermal-hydraulic (T&H) characteristics. From this information the Plant Thermal Hydraulic design condition (the basis of component sizing) was selected. Performance calculations, based on specified component sizing, and uncertainty in component performance, have subsequently been made. The temperatures from these calculations form the basis for a - priori fuel burnup projections and estimates of "stretch" power operational capability. It should be emphasized that PSAR accident analysis and structural evaluations of permanent plant components are based on worst case plant temperatures and flows - not on the expected temperatures and flows from the performance calculations.

Reference

J. D. Mangus, P. B. Deegan, "Selection of the Design Conditions for the Clinch River Breeder Reactor Plant," WARD-D-0027, Feb 1974.

A.60 MINX

MINX calculates fine-group averaged infinitely dilute cross sections, self-shielding factors, and group-to-group transfer matrices from ENDF data. MINX incorporates and improves upon the resonance capabilities and the high-legendre-order transfer matrices of existing codes. Group structure, Legendre order, weight function, temperature, dilutions, and processing tolerances are all under user control.

Availability

MINX as released in September 1976, is currently available at the Los Alamos Scientific Laboratory (LASL) computer facility in Los Alamos, New Mexico.

Verification

MINX will become an integral part of the final design phase for CRBRP. It will be used in the reference design method for calculating neutron cross section data for both the plate geometry critical experiments and the pin geometry CRBRP core. This cross section data will be used in nuclear analysis computer codes as part of the reference calculational technique to analyze critical experiments and establish bias factors and uncertainties for criticality, reactivity coefficients, control rod worths, reaction rates and other neutronic parameters. These bias factors and uncertainties will then be applied to the analysis of the CRBRP core, again using reference design methods employing cross sections generated, in part, by MINX. This proposed verification scheme is similar to the current scheme which used the ETOX, XSRES-1DX and ANISN codes to generate cross section data.

Application

MINX generates pseudo-composition independent multigroup libraries in the standard Committee on Computer Code Coordination (CCCC)-III interface formats for use in the design and analysis of nuclear systems. The output from MINX is used as input to the SPHINX code.

Reference

C. R. Weisbin, P. D. Soran, R. E. MacFarlane, D. R. Harris, R. J. LaBauve, J. S. Hendricks, H. E. White and R. B. Kidman, "MINX, A Multigroup Interpretation of Nuclear X-Sections from ENDF/B," LA-6486-MS, September 1976.

A.61 MRI/STARDYNE

The STARDYNE Analysis System is a series of compatible computer programs for analyzing the finite element method linear elastic structural models for a full range of static and dynamic input conditions. The static capability includes the calculation of structural deformations and members loads/stresses caused by an arbitrary set of thermal and/or applied loads. Prescribed displacement vectors can be used as input to compute resulting internal deformations, loads and stresses. The dynamic capability includes normal mode response analyses for a wide range of loading conditions including transient, steady state harmonic, random and shock spectra. Dynamic response results can, in general, be presented as structural deformations (displacements, velocities, or accelerations), and/or internal member loads/stresses.

Availability

STARDYNE (CDC-8400-2500) had been available on the CDC 7600 computer of Lawrence-Berkeley Laboratory.

STARDYNE is also available on the CDC-Cybernet System.

Verification

Documentation of verification of the STARDYNE computer code per SRP Section 3.9.1.II.2.c primary and intermediate can be found in the reference.

Application

The STARDYNE code is being used to perform global dynamic analysis of the pumps including preliminary normal modes analysis and seismic and rotor imbalance response analyses.

STARDYNE has also been used for static and seismic analysis of structures.

Reference

MRI/STARDYN Finite Element Demonstration Problems, Document No. 84002500, Control Data Corporation, Minneapolis, Minn.

A.62 NAPALM

The NAPALM program is used to analyze nozzles or penetrations subjected to mechanical loads and to find the magnitude and location of the maximum stress intensity.

The program analyzes at specified axial locations on the penetration. The calculations are based on the familiar strength of materials formulation.

Availability

This program is available on the IBM 370, model 165 computer of the Chicago Bridge and Iron Company.

Verification

Verification of this program is achieved through the use of the following problem:

> The problem considered is a 16 1/16 O.D. by 13 7/8 inch I.D. nozzle, loaded by nozzle loads and thermal sleeve loads. Nozzle loads are applied at the safe end and thermal sleeve loads are applied 5 inches from the safe end. An internal pressure of 1000 psi is also applied to the nozzle.

The computer results are compared to analytical results. As shown Table A.62-1 the agreement between the computer and analytical solutions is almost exact. The values presented are the maximum stress intensities.

Application

NAPALM will be used in the design of the penetrations for the CRBRP Head Assembly.

Reference

"CRBRP Design and analysis of Head Assembly, Computer Programs and Program Control" Report 54500-4.2.3 Revision 0, Chicago Bridge and Iron, Oak Brook, Illinois, dated March, 1976.

Table A.62-1 NAPALM Results

Quantity	Units	Outside Surface	Inside Surface	Membrane Surface	Analytical Results Outside Surface	Analytical Results Inside Surface	Analytical Results Membrane Surface
Maximum Stress Intensity	PSI	14565.	14409.	14481.	14564.	14409.	14481.
THETA	Degrees	324.	324.	324.	324.	324.	324.
FX	LBS	10000.	10000.	10000.	10000.	10000.	10000.
FY	LBS	10000.	10000.	10000.	10000.	10000.	10000.
FZ	LBS	-8000.	-8000.	-8000.	-8000.	-8000.	-8000.
MX	IN-LBS	-400000.	-400000.	-400000.	-400000.	-400000.	-400000.
MY	IN-LBS	400000.	400000.	400000.	400000.	400000.	400000.
MZ	IN-LBS	-700000.	-700000.	-700000.	-700000.	-700000.	-700000.
LONG. STRESS (PRESSURE)	PSI	0.	0.	0.	0.	0.	0.
LONG. STRESS (AXIAL LOAD)	PSI	-212.	-212.	-212.	-212.	-212.	-212.
LONG. STRESS (BENDING)	PSI	-6086.	-5325.	-5705.	-6086.	-5325.	-5706.
SHEAR STRESS (FORCES + TORSION)	PSI	3446.	-3032.	-3239.	3446.	-3033.	-3240.
CIRC. STRESS (PRESSURE)	PSI	6533.	7533.	7033.	6533.	7533.	7033.
S1	PSI	7400.	8203.	7798.	7400.	8203.	7798.
S2	PSI	-7165.	-6206.	-6682.	-7164.	-6206.	-6683.
S3	PSI	0.	-1000.	-500.	0.	-1000.	-500.

A.63 NASTRAN

NASTRAN is a finite-element program for static and dynamic structural analysis developed by NASA. It uses the following structural modeling elements: beams and rod, plates (anisotropic material properties optional) shear and twist panels, conical shells, solid polyhedrons and solid rings.

Availability

The advanced proprietary MacNeal - Schwendler version of NASTRAN is available at the UNIVAC 1108 computer of Information Systems Design, Santa Clara, California. Since it is on a commercial data center, the running cost is based on the computer time plus royality.

Verification

It is recognized and widely used in industry with a sufficient history of successful applications to justify its validity per SRP 3.9.1, Section II.2.a. Validation is documented in Reference 1.

Application

It will be used for structural analysis and/or thermal analysis on some structures and/or components.

Reference

MacNeal, R. H., "The NASTRAN Theoretical Manual," NASA-SP-221, September 1970.

A.64 NICER (Westinghouse Proprietary)

The NICER code is used to calculate detailed thermal-hydraulic performance of fuel and blanket assemblies. The code has a modular structure with subroutines or groups of them performing specific calculations. The most important are:

- Calculation of axial and radial temperature profile in any rod of fuel and blanket assemblies; subchannel coolant temperature profile calculated by subchannel analysis codes (e.g., CØTEC) is adopted as boundary condition;
- Calculation of fission gas plenum pressure; a fission gas release model similar to the one in LIFE-III is featured in NICER;
- Calculation of assembly mixed mean temperatures;
- Calculation of rod bundle pressure drop on the basis of the Novendstern correlation (Ref. 2, CØTEC code); more detailed assembly pressure drop correlations are featured in CATFISH.

All the above calculations are performed through lifetime and at various uncertainty levels of confidence. This, coupled with the very fast running time, makes the code very valuable for core-wide mapping and lifetime evaluations. Additionally, NICER has the capability to evaluate the value of those hot channel/spot factors which are functions of the axial position and/or the operating conditions of the assembly. Both systematic and statistical hot channel factors are considered and combined at a user's specified level of confidence.

Fuel rod temperatures are also calculated as input to transient and safety analyses, while for steady state power-to-melt calculations NICER provides the boundary conditions to detailed LIFE analyses. Empirical models of the fuel/gap behavior derived from LIFE and calibrated against early-in-life test data, such as the HEDL P-19, are included in NICER.

Finally, the code features a large number of options to bypass calculations of no immediate interest, to vary calculational models, to select desired correlations or to choose pertinent material properties.

Availability

The NICER code is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

Since NICER is a modular type code, separate validation is applied to each calculational module, as follows:

- Coolant temperature calculations: since this is input to NICER from the subchannel analysis codes, the verification discussed under CØTEC, CØBRA and THI-3D applies;

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- Cladding temperature calculations: these calculations are straightforward application of Fourier one-dimensional heat transfer law, employing verified material thermal properties;

- Fission gas release calculations: since NICER adopts models similar to LIFE, verification under LIFE applies to NICER. In addition, NICER fission gas release calculations have been directly compared against EBR-II experimental data, showing very good agreement;

- Calculations of mixed mean temperature: this is simply derived from an energy balance/enthalpy rise correlation, thus hand checks are sufficient;

- Bundle ΔP calculations: NICER calculated values are being verified against pertinent experimental data, chiefly FFTF and CRBRP core assembly testing. Also see CATFISH verification;

- Fuel temperature calculations: since NICER adopts LIFE simplified models, LIFE verification applies, provided that NICER/LIFE consistency is satisfactorily verified.

- Hot spot factors: most of the hot spot factors are input to NICER, thus, they must be verified independently. This will be done by comparison against experimental data (e.g., wire wrap effect, see FATHØM-360, nuclear uncertainties and inlet flow maldistribution), specified environmental conditions (e.g., effect of tolerances and operating conditions) and analyses performed with verified codes (e.g., nuclear uncertainties and flow distribution calculational uncertainties). Regarding the hot spots directly calculated in NICER, hand calculations and critical review of bases will be performed.

Application

The NICER code is used to predict the steady state thermal performance of fuel and blanket rods.

References

1. M. D. Carelli, C. W. Bach and R. A. Markley, "Analytical Techniques for Thermal-Hydraulic Design of LMFBR Assemblies", Trans. Am. Nucl. Soc., 17, pp. 423-424.

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A.65 NONSAP

NONSAP is a finite element structural analysis program for the static and dynamic response of nonlinear systems. The program is an in-core solver. The capacity of the program is essentially determined by the total number of degrees of freedom in the system. However, all structure matrices are stored in compacted form, i.e. only nonzero elements are processed.

The system response is calculated using an incremental solution of the equations of equilibrium with the Wilson θ or Newmark time integration scheme. Before the time integration is carried out, the constant structure matrices, namely the linear effective stiffness matrix, the linear stiffness, mass and damping matrices, whichever applicable, and the load vectors are assembled and stored. During the step-by-step solution the linear effective stiffness matrix is updated for the nonlinearities in the system. Therefore, only the nonlinearities are dealt with in the time integration.

Availability

NONSAP (UC-SESM 74-3, February 1974) is available on the CDC 7600 computer of Lawrence Berkeley Laboratory. The version was released in February, 1974.

Verification

Verification of NONSAP per SRP Section 3.9.1.II.2.b is available in References 1 and 2. A comparison of results is shown in Figure A.65-1.

Application

NONSAP will be applied to structures in which the nonlinear response is significant.

References

(1) Bathe, K. J., Ozdemir, H., and Wilson, P. L. "Static and Dynamic Geometric and Material Nonlinear Analysis," Structures and Materials Research Report UC SLSM 74-4, University of California, Berkeley, Feb. 1974.

(2) Bathe, K. J., Wilson, E. L., and Iding, R. H., "NON-SAP- A Structural Analysis Program for Static and Dynamic Response of Nonlinear Systems," Structure and Materials Research Report, UC SESM 74-3, University of California, Berkeley, Feb. 1974.

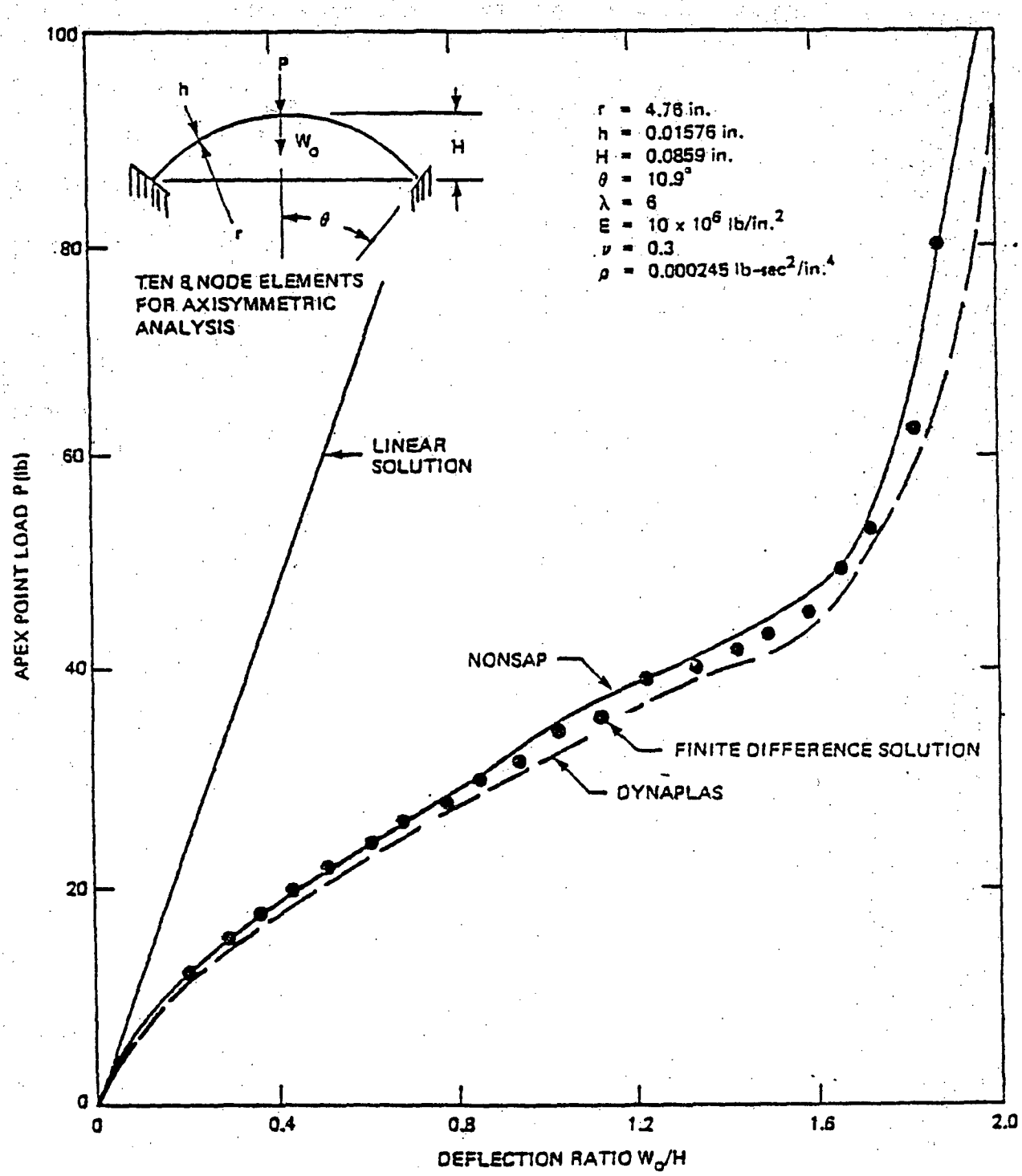

Fig. A.65-1 *Comparison of NONSAP with the DYNAPLAS Computer Code for a Spherical Shell*

A.66 ØCTØPUS

The ØCTØPUS code optimizes the allocation of reactor coolant flow among fuel and blanket assemblies to satisfy a priori all design criteria such as: a) burnup/lifetime and transient limitations which are expressed through equivalent limiting cladding temperatures, and b) mixed mean temperature level and thermal gradients, through equalization and minimization of the fuel and blanket assemblies mixed mean exit temperature. To accomplish this, strain, cladding damage and transient equivalent limiting temperatures are established for reactor fuel and radial blanket assemblies and coolant is allocated according to the most restrictive of these limitations. The code takes into account the power generation fraction, axial and radial nuclear peaking factors, axial power distributions, and hot spot factors calculating the minimum flow in each assembly required to satisfy the lifetime and transient constraints. Assemblies are then grouped into a specified number of orificing zones. All the possible combinations of N assemblies in n orificing zones are examined by ØCTØPUS and the configuration yielding the minimum amount of total core flow required is chosen. The amount of total core flow thus allocated is compared to the available amount; if an excess flow exists, this is distributed among the assemblies to minimize exit temperatures and gradients.

Availability

The ØCTØPUS code is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

ØCTØPUS calculations of those assembly thermal-hydraulic parameters (i.e., assembly flow rate, rod maximum cladding temperature and axial location of maximum) which define the core orificing scheme are being verified against parallel calculations performed using the CØTEC (A.14) and NICER (A.64) codes. Excellent agreement was found as shown in Table A.66-1, which reports sample calculations for the 9 zones orificing scheme of the CRBRP homogeneous core.

Application

The ØCTØPUS code is used to determine the optimum distribution of reactor coolant flow among fuel and blanket assemblies accounting for all relevant constraints.

Reference

1. M. D. Carelli, A. J. Friedland, C. W. Bach and R. A. Markley, "An Optimized Method for Orificing LMFBR Cores", Trans. Am. Nucl. Soc., 26, pp. 437-438 (June 1977).

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TABLE A.66-1

COMPARISON OF ØCTØPUS WITH PARALLEL CØTEC/NICER CALCULATIONS

	FLOW ORIFICING ZONE								
	ZONE 1	ZONE 2	ZONE 3	ZONE 4	ZONE 5	ZONE 6	ZONE 7	ZONE 8	ZONE 9
NICER - SELT, °F	1180.70	1199.27	1235.75	1227.03	1241.22	1275.50	1271.00	1260.70	1269.98
ØCTØPUS FINAL MAXIMUM CLADDING TEMPERATURE, °F	1180.86	1198.91	1235.46	1227.03	1241.22	1275.48	1270.98	1261.06	1269.96
NICER AXIAL HEIGHT WHERE MAXIMUM CLADDING TEMPERATURE WAS FOUND, INCHES	50.25	50.25	50.25	50.25	50.25	42.00	42.00	46.00	48.00
ØCTØPUS AXIAL HEIGHT WHERE MAXIMUM CLADDING TEMPERATURE WAS FOUND, INCHES	50.25	50.25	50.25	50.25	50.25	42.00	42.00	46.00	48.00
CØTEC ASSEMBLY FLOW RATE, LBM/HR	190255.0	163880.0	141700.0	134785.0	114050.0	58965.0	54785.0	33375.0	17485.0
ØCTØPUS ITERATED ASSEMBLY FLOW RATE, LBM/HR	190560.9	164395.1	142122.6	135093.7	114307.4	58895.86	54869.26	33524.68	17594.91
ABSOLUTE PERCENT DIFFERENCE, $\frac{\dot{m}_{ØCTØPUS} - \dot{m}_{NICER}}{\dot{m}_{NICER}}$ IN ASSEMBLY FLOW RATE	0.160%	0.314%	0.298%	0.229%	0.226%	0.117%	0.153%	0.440%	0.628%

A.67 ORIGEN

ORIGEN is a point depletion code which solves the equations of radioactive growth and decay for large numbers of isotopes with arbitrary coupling. The code uses the matrix exponential method to solve a large system of coupled, linear, first-order ordinary differential equations with constant coefficients. The general nature of the matrix exponential method permits the treatment of complex decay and transmutation schemes. A library of nuclear data, including half-lives and decay schemes, neutron absorption across sections, fission yields, disintegration energies, and multigroup photon release data is used.

Availability

The ORIGEN code is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center. The version currently in use was released from ORNL in May, 1973.

Verification

Code verification was performed by the originator of the code (ORNL). A test case transmitted to ARD from ORNL as part of the code package and run on ARD's CDC system produces results consistent with ORNL. Comparisons versus RIBDII code predictions of similar quantities provide further verification.

Application

ORIGEN, in its CRBRP application, has been used to predict transuranium and transactinide inventories in the CRBRP fuel and blanket assemblies. This information is used to predict inherent neutron source levels for use in flux monitor studies and fuel handling system shielding.

Reference

M. J. Bell, "ORIGEN - The ORNL Isotope Generation and Depletion Code," ORNL-4628, May 1973.

A.68 PDA (Proprietary General Electric)

The Pipe Dynamic Analysis Program (PDA) is used to determine the response of a pipe subjected to thrust forces occurring after a pipe break. To limit the pipe movement, a restraining device may act to inhibit the motion following the break. Pipe and restraint movements are modeled using non-linear differential equations. The solution of the equation is obtained using a digital computer program. The program analyzes the thrust force applied to the end of the pipe and the restraining forces (from the restraint, structure, and pipe) tending to maintain the position of the pipe. The program goes through 5 modes of calculation summing the moments generated by the applied forces and computing angular acceleration of the pipe.

Availability

PDA was developed by General Electric Co. - Nuclear Energy Division, and is available at the GE computer center at San Jose, CA.

Verification

General Electric Co. contracted Nuclear Services Co. (NSC) to perform an independent analysis of a recirculation system piping arrangement. The results, as reported in the Reference, were submitted by GE to NRC as part of GESSAR. The verification is in accord with SRP Section 3.9.1.II.2.c.

Application

PDA will be used in analyzing the SGS and SGAHRS high energy water/steam piping to determine the pipe deflections resulting from postulated pipe breaks.

Reference:

Nuclear Service Corporation Report No. GEN-02-02, "Final Report Pipe Rupture Analysis of Recirculation System for 1969 Standard Plant Design", May 1973.

A.69 PERT-V

PERT-V is a two-dimensional, perturbation theory code designed to calculate reactivity coefficient and material worth distributions, the effective delayed neutron fraction, the neutron generation time, and the inhour/ k conversion factor using forward and adjoint fluxes output from neutronics codes such as XSRES-W1DX and W-2DB. First order perturbation theory, based on the multigroup diffusion model, is used to calculate reactivity coefficients and each component of the perturbation equation at every mesh point in the solution.

Availability

The PERT-V code is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

PERT-V has been used to calculate the spatial distribution of various reactivity coefficients. Zero Power Plutonium Reactor (ZPPR) critical experiment measurements and measurements made in other fast reactor cores were analyzed with this code to assess its accuracy. The results obtained from these comparisons are fully described in subsection 4.3.2.3 of the PSAR. Further verification will be performed based on Engineering Mock-up Critical data. A set of test problems, specifically designed for code verification, will be developed.

Application

Specifically, the PERT-V code was used to calculate the spatial distribution of the Doppler and Sodium Void reactivity coefficients. Experiments performed in the Southwest Experimental Fast Oxide Reactor (SEFOR) were analyzed with reference design techniques, including PERT-V, to determine the accuracy of Doppler coefficient calculations in a typical LMFBR environment. The ZPPR-3 Modified series of critical experiments were analyzed in detail to assess the uncertainties associated with calculations of sodium void worth in LMFBR cores. PERT-V, which treats the sodium removal as the perturbation to the cross sections,was used to obtain the spatial distribution of the sodium void worth in two-dimensional (R-Z) calculations of these experiments. The same techniques, with the appropriate bias factors and uncertainties, are applied to calculations of these reactivity coefficients in the CRBRP core.

Reference

R. W. Hardie and W. W. Little, Jr., "PERT-V: A Two-Dimensional Perturbation Code for Fast Reactor Analysis," BNWL-1162, September 1969.

A.70 PIP

PIP is a digital computer program which provides a means of predicting pressure losses for a fluid being forced through a piping system. The methods employed in this program provide a code applicable to a wide range of Reynolds' number with varying degrees of pipe roughness. The program is flexible, having the capacity to account for the following:

Any fluid may be used, provided its density and viscosity is known as a function of temperature.

Any pipe material may be used. The expansion coefficients as a function of temperature should be known.

Any three fluid operating temperatures may be used in one problem.

Any six fluid flow rates may be used in one problem.

The program accounts for horizontal and vertical piping runs through the following piping configurations:

- straight pipe
- gradual enlargements
- gradual contractions
- 90° elbows
- 45° elbows

Provisions are included for equipment in the system. The pressure losses and fluid hold-up in the equipment should be known.

A maximum of 75 pipe sections per problem is permitted.

Availability

PIP is available only as a fortran card deck at ARD for the CDC-7600 computer.

Verification

The code which determines the solution of closed form arithmetic expression has been extensively checked through hand calculations.

Application

PIP is used to determine CRBRP PHTS piping pressure losses.

Reference

J. Wasko, "Pressure Losses in Piping Systems Code Utilization Description," Westinghouse Electric Corp., Advanced Reactors Division, Madison, Pa., WCAP-7143, December 1967.

A.71 PLAP

The PLAP code provides a solution of the plastic response of a pipe to transient overpressure and the degradation of the peak pressure with travel through the pipe. It is a marching solution, through both time and space, of the equations of dynamic equilibrium of the pipe wall and the compressed fluid, at successive instants, for a pressure pulse traveling the length of pipe. Input parameters include the properties of both the pipe and the fluid, plus the initial pressure-time history of the pulse in the form of a fitted polynomial curve. The code output includes stress, strain, acceleration and radial velocity of the pipe wall, plus the pressure-time history of the degraded pulse.

Availability

The program has been installed in the GE Time Share System Library at San Jose. It is a routine written in FORTRAN and can be installed with any compatible Fortran Assembly program. A source deck of card input can be obtained by applying to GE-FBRD at 310 DeGuigne, Sunnyvale, Calif.94086.

Verification

The program has been verified per SRP Section 3.9.1.II.2.c, by comparison with test results from Stanford Research Institute reported in HEDL-SRI-1, UC-79p "Simulation of a Care Disruptive Accident in a Fast Flux Test Facility", A. L. Florence and G. R. Abrahamson, May 1973. The verification is documented in the reference.

Application

PLAP is suitable for determining the ability of pipe to resist transient overpressure pulses that plastically deform the pipe, the degradation of peak pressure from the energy absorbed in the plastic deformation as a function of the travel distance of the pulse, and the deformation pattern resulting in the pipe from such passage.

Reference

"Plastic Response of Pipe to Transient Overpressure", General Electric Co., Sunnyvale, California, GE-NEDC-14028, December 1974.

A.72 PUMA-8

PUMA-8 is a two-dimensional, triangular geometry, one-group burnup code for use in fast reactor radial blanket fuel management analyses. PUMA uses start and end of cycle fluxes and gamma heating by mesh interval from the W-2DB two-dimensional diffusion theory, or similar, code. Power and burnup are calculated in the one-group diffusion approximation using spectrum-weighted fission and absorption cross sections from W-2DB. The fluxes, gamma heating, and cross sections are deduced for each fuel rod in the blanket by means of a two-dimensional quadratic least squares fit. The code is then used to predict the power, burnup, isotopic fission and capture rate and plutonium inventory histories throughout the lifetime of each rod in the radial blanket. An estimate of the variations in axial peaking factor with burnup and rod position is also provided.

Availability

The PUMA-8 code is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

Verification of the PUMA blanket power and burnup results will be accomplished by comparison with higher-order diffusion-eiganvalue calculations. These comparisons will include pin-by-pin fuel depletion and plutonium buildup, power distributions, and isotopic fission rates. The diffusion calculations in turn, are verified by comparison with critical experiments in ZPPR.

Application

PUMA-8 is used to compute radial blanket power, isotopic fission and capture rate, burnup and material inventory histories by blanket pin. The code produces design oriented data for blanket thermal-hydraulic and decay heating analyses. It is also used to investigate blanket fuel management effects of assembly lifetime (residence time), shuffling and rotation.

A.73 QAD

QAD is a gamma ray point kernel code designed to calculate radiation levels at detector points located within or outside a complex radiation source geometry describable by a combination of quadratic surfaces. The code evaluates the material thicknesses intercepted along the line-of-sight from the source point to the detector point. These material thicknesses (or path lengths) are employed in attenuation functions to calculate the gamma flux, dose rate and heating rate at the detector point. The attenuation function uses expontential attenuation along with a build-up factor (Capo's method).

Availability

The QAD code is available from the Argonne National Laboratory Code Center and the Radiation Shielding Information Center, and is used on the CRBRP project by Burns and Roe.

Verification

The program has been used and checked widely by the industry to justify its validity per SRP 3.9.1, Section II.2.a. Results have been found reasonable when compared with other point kernel shielding programs.

Applications

QAD is used to determine bulk shielding wall thicknesses to meet the CRBRP radiation zone requirements.

Reference

R. E. Malenfant, "QAD: A series of Point Kernel General Purposes Shielding Programs", LA 3573 October 1966.

A.74 RIBDII

RIBD is a grid processor which calculates isotopic concentrations resulting from two fission sources (for example, Pu-239 and U-238) with normal down-chain decay by beta emission and isometric transfers and inter-chain coupling resulting from n-gamma reactions. The calculations follow the irradiation history through an unlimited number of step changes of unrestricted duration and variability including shutdown periods, restarts at different power levels and/or any other level changes. Changes from the eariler versions of the RIBD code are: the expansion to include up to 850 fission product isotopes, input in the user-oriented NAMELIST format, and run-time choice of fissionable isotopes from a library of nuclear data. The library that is included in the code package contains yield data for 818 fission product isotopes for each of fourteen different fissionable isotopes, together with fission product transmutation cross sections for fast and thermal systems.

Availability

The RIBDII code is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center. The version currently being used was released from Hanford Engineering Development Laboratory (HEDL) in January, 1975.

Verification

Code cerification was performed in accord with SRP Sections 3.9.1.II.2.b and c by the originator of the code as indicated in the reference. A test case transmitted to ARD from HEDL as part of the code package and run on ARD's CDC system produces consistent results. Verification of results between the codes RIBD and RIBDII was performed at ARD by running test cases using both codes. Input data libraries are ENDF/B-IV information with verification provided by comparison to experimental measurements of fission product energy release.

Application

RIBDII application in CRBRP is for calculation of fission product inventories for CRBRP fuel and blanket assemblies and for operating power histories defined by fuel management and power performance analyses. In addition, fission product yield data for 14 fission reactions with isotopes of maximum plutonium and thorium (function of incident neutron energy) are available to calculate individual fission product isotopic activities (curies), concentrations (gam-atoms), and beta and gamma decay powers (megawatts). In decay power analysis for CRBRP, RIBDII is used to calculate time dependent decay energy release data for fission of principal fissionable isotopes. This data is used in the S-4 code to predict fission product decay power full power, lead follow, and transient conditions.

Reference

D. R. Marr, "A User's Manual for Computer Code RIBDII - A Fission Product Inventory Code," HEDL-TME-75-26, January, 1975.

A.75 S-4

S-4 is a decay energy release rate code which solves for the time dependent energy release rate due to fission products and transuranium (Np-239, U-239) isotopes. The code employs a convolution of a reactor power/time or fission rate/time history with empirically derived equations for decay energy release from fission product isotopes for various fissionable isotopes to calculate fission product decay power. A solution of the coupled differential equations for U-239 and Np-239 buildup and decay is used for transuranium isotope decay power. Total decay power at user specified times during reactor operation and after shutdown can be computed. Decay energy release data for fissionable isotopes is based on experimental data or evaluated nuclear data file (ENDF/B-IV) as computed by the RIBDII code.

Availability

The S-4 code (S4M version, May 1977; Traceable to Ref. 1) is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

The S-4 code and data have been verified by comparison to measured data and the ORIGEN and RIBDII codes. ENDF/B-IV Kernel data is currently being verified by HEDL in an experimental program.

Application

The S-4 code is used for calculating fission product and transuranium element decay power for shutdown conditions out to several years after shutdown. Decay energy release rate data have been incorporated into the code to calculate individual fissionable isotope decay powers. Either powers or fission rates may be input to calculate fission product decay power for U-235, U-238, and Pu-239. Transuranium element decay is calculated using the U-238 capture rate to provide U-239 and Np-239 decay power. Exposure histories can be input for each chosen operating time point, or as Beginning-and End-of-Cycle values such that S-4 will assume the time step characteristics to be linear with time. Individual zone decay powers by isotope can be calculated and summed over a total system within the code. Output values can be expressed as fractions of operating power, percents, or in powers.

References

1. McGinnis, C. A., Tomlin, D. O., Disney, R. K., "CRBRP Decay Power Analysis," WARD-D-0090, Apendix A, January 1976.

2. Westinghouse ARD Internal Documentation - Nonproprietary.

A.76 SAP-IV

SAP-IV is a general purpose, linear elastic, static and dynamic finite element computer code. This code is used primarily to perform thermal expansion, dead weight, and seismic analysis of HTS piping system. Options are available for dynamic evaluations, modal extraction, and transient response. In the latter cose, the program uses either a stable numerical scheme or mode super position approach to evaluate the response time history.

Availability

The version of SAP IV, released in April, 1974 is currently available both on the Honeywell 6000 computer of Nuclear Energy Systems Divisions, General Electric Company, San Jose, California, on the CDC 7600 computer of Lawrence Berkeley Laboratory, Berkeley, California, and on the IBM 370/165 computer of the Chicago Bridge and Iron Company.

Verification

The description of the program is well documented in Ref. 1. The Honeywell version has been verified against PIPDYN, ADL-PIPE, and ANSYS (WESTDYN) for some bench mark problems. SAP IV is recognized and widely used in industry with a sufficient history of successful applications to justify its validity per SRP 3.9.1, Section II.2.a.

Application

SAP IV will be utilized for the static and dynamic analyses of piping systems, dump tanks, sodium pumps, and the reactor vessel closure head.

Reference

Bathe, K. J., Wilson, E. L., and Peterson, F. E., "SAP IV - A Structural Analysis Program for Static and Dynamic Response of Linear Systems, Report No. EERC 73-11, University of California, Berkeley, California, June 1973, Revised April, 1974.

A.79 SCAP-BR

SCAP-BR is the modified version of SCAP code incorporating the capability of analyzing gamma radiation scattering through multi-legged penetrations. SCAP-BR is a point kernel integration code employing an angle/energy dependent multiple scatter method in a complex geometry. SCAP-BR calculates the radiation level at detector points located inside or outside of a complex scattering geometry. The geometry is described by zones bounded by intersecting quadratic surfaces. The attenuation of the gamma rays through shielding material is approximated by an exponential function with buildup factors.

The energy dependent cross sections for photoelectric and pair production of gamma-rays are computed by using the built-in data library. Scattering cross sections are calculated by using the Klein-Nishina Compton scattering formula.

The gamma-ray source geometry can be either an isotropic point source or volumetric source. The code calculates both direct and scattered gamma-ray flux at each detector point. The direct portion of the flux, from a volumetric source, is calculated based on the same method as employed in QAD. A volumetric source is transformed into an effective point source which yields the same dose rate contribution at the duct entrance as the volumetric source for calculation of the scattered flux.

The multiple scattering is handled by a repeated process of the single scatter method through successive scatter regions. The scattered gamma rays at each scattering point in a scattering region contribute a dose rate to a detector point. At the same time, these scattered gamma rays in each region with their degraded energies are grouped and collapsed into psuedo source points. These pseudo-source points with their collapsed source strengths will "see" the next scattering region for the next scattering region scattering analysis. The process is repeated for each region to account for the multiple scattering efforts.

Availability

SCAP-BR is available at Burns and Roe, Atomics International and Westinghouse Advanced Reactors Division on CDC6600 and 7600 computers.

Verification

SCAP-BR has been verified by comparing its dose rate predictions with known experimental data and Monte Carlo results. The verification of SCAP-BR is documented in the reference.

Application

SCAP-BR is used to calculate the gamma ray scattering through penetration and ducts between radioactive and non-radioactive cells to meet the radiation zone criteria for cells and areas in the Nuclear Island.

Reference

(1) Byoun, T. Y. Babel, P. J., Dajani, A. T. , "Computer Code SCAP-BR" Description, January 1978.

A.80 SETS

The SETS code calculates the steam/water and bulk metal temperature distribution and steam/water side pressure throughout the CRBRP steam generator using the sodium and water side inlet boundary conditions as input. It was modified from the TAP (Reference 1) program to be used exclusively for the CRBRP steam generator. It is capable of performing both steady-state and transient analyses and can be used for analyzing both single-phase and two-phase flow on the water/steam side.

Availability

SETS (Ref. 2) is available on the IBM 360 computer of Rockwell International.

Verification

The code has been verified by comparing the results with those of the DEMO code. The agreement between the two was found to be very good (Reference 3).

Application

The SETS code is used to generate the steady-state and transient boundary conditions for various components of the steam generator. The boundary conditions consist of steam/water and sodium temperatures throughout the length of the steam generator as well as their respective heat transfer coefficients. These boundary conditions are then used as input to the TAP thermal model of the individual component in calculating time dependent temperature distribution in the component.

Reference

1. E. Moody, "Thermal Analyzer Program Revision (TAP-4F)," TI-099-411-015, AI, December 23, 1975

2. C. Tan, "SETS User's Manual," N036TI620070, AI, October 14, 1976

3. Von Arx, Allan, "SETS Code Verification," N036TI620-042, AI, April 30,1976

A.81 SNAP (General Electric Properietary)

SNAP determines elastic stresses and displacements in assemblies of axisymmetrically loaded shells of revolution (cylinders, cones, discs, toroidal rings of rigid cross section), all with spanwise variations of temperature and linear thermal gradient. SNAP additionally provides for pinned and sliding joint configurations together with unusual shell elements (cutouts) for which equilibrium and compatibility equations can be written. The SNAP program has been used to model the inner portion of the inlet nozzle and a portion of the driveline. It will be utilized whenever the required analyses lend themselves to a solution with the SNAP code.

Availability

SNAP (May, 1971) is available on the Honeywell 6000 computer system of General Electric in San Jose.

Verification

SNAP which is also known as MULTI-SHELL III is the improved version of the computer code MULTI-SHELL II. MULTI-SHELL code has been extensively used over the last two decades within the various divisions of General Electric Company. There have been a number of physical tests which verified the validity of the code per SRP Section 3.9.1.II.2.c. The documention on these verifications are listed in the reference.

Application

SNAP is used in the type of the structure which is axisymmetrical to the axis of revolution with non-zero radius, such as the Sodium Dump Tank and the Reaction Products Separator Tank.

Reference

"SNAP-Improved Computer Program For the Analysis of SHELL of Revolution with Axisymmetric Loading", General Electric Technical Information Series No. R66FP0171, Nov. 1966.

A.82 SOFIRE-II

SOFIRE-II is a digital computer code[1] which solves a set of theoretical equations describing the simultaneous process of heat and mass transfer by the finite difference method. The physical system is simulated by a nodal network in which the nodes are connected to each other along a heat transfer path by admittances (the reciprocal of the thermal resistance). Each node has a thermal capacity equal to that of the physical counterpart represented by the node. The temperature assigned to or calculated by each node represents the temperature at the centroid of the corresponding elements in the physical system.

Two versions of SOFIRE-II have been written to simulate a pool fire in a single containment volume and in an interconnected double cell.

Availability

SOFIRE-II is available on the Honeywell 6000 computer systems of General Electric in San Jose and on the CDC 7600 computer of Lawrence Berkeley Laboratory.

Verification

Experimental verification using a large vessel at sodium temperatures about 1000°F, together with results of small open air pool fires provided sufficient understanding of the sodium burning process that SOFIRE-II can be anticipated to conservatively predict the pressures resulting from potential LMFBR accidents. Verification of the SOFIRE-II code was also discussed with NRC on March 5, 1976. The summary of this meeting is presented in reference 2. In addition, a discussion of the code and its verification were presented in the response to NRC question 001.239 and associated references.

Application

The SOFIRE-II code is used to describe the pressure-temperature history in inerted cells following a postulated sodium spill. The theoretical model has been developed to compute 1-cell and 2-cell pool fire cases. The basic equations of the Code are not restricted to any specific size of the sodium pool and cell.

Reference

1) P. Beiriger and others, "SOFIRE-II User Report," AI-AEC-13055, March 1973.

2) Letter, P. S. Van Nort to R. S. Boyd, "Summary of Meeting held between CRBRP Project and NRC to Discuss CACECO and Sodium Fires and SPRAY Codes used in CRBRP Analysis," March 10, 1976.

A.82.A SPCA

SPCA (Spray-Pool Combustion Analysis) is a digital computer code which solves a set of theoretical equations describing the simultaneous process of heat and mass transfer to provide cell gas temperature and pressure, concrete temperature and cell gas temperature time histories. The physical system is simulated by a nodal network in which the nodes are connected to each other along a heat transfer path by admittances (the reciprocal of the thermal resistance). Each node has a thermal capacity equal to that of the physical counterpart represented by the node. The temperature assigned to or calculated for each node represents the temperature at the centroid of the corresponding element in the physical system.

The model computes the heat generation from sodium or NaK spray burning and pool burning. The spray and pool burning thermal effects are combined into a single analysis. The Thermal Analyzer Program (TAP-4F) (A.89) is used for the integration of the heat transfer equations. The thermal effects of spray and pool burning are simulated by the addition of heat at the appropriate node of the TAP nodal model representing the cell or building structure in which the fire occurs.

Availability

SPCA is under development at Rockwell International, Energy Systems Group. the code is programmed for the IBM 370 computer system.

Verification

Verification of the equations used in-the-code has been completed (Ref.). Verification consisted of verifying by hand calculation, specific machine-calculated parameters such as the spray burning rate, the pool burning rates, the aerosol generation rate, and the gas venting rate. Mass and energy conservation checks were also made. Comparisons between SPCA calculations and SOFIRE-II and SPRAY-III code calculations will be documented. Experimental validation of the SPCA calculations will consist of pre and post-test comparisons with the measured parameters of planned large-scale sodium fire tests. For a discussion of sodium fire tests, see Section 1.5.2.11.

Application

The SPCA code is used for the analysis of the consequences of sodium/NaK leak (fire) accidents in air-filled cells.

Reference

AI specification N099TI260013 Revision A, "Spray Pool Combustion Analysis (SPCA) Code".

A.83 SPECEQ/SPECUQ

These programs compute response spectra from earthquake accelerograms digitized at equal (SPECEQ) or unequal (SPECUQ) time intervals. The generated response spectra represent the maximum responses of a damped single degree of freedom system to the specified earthquake ground acceleration. An exact analytical solution to the governing differential equation is used in the computation for successive linear segments of the excitation.

The programs are written in FORTRAN IV and are dimensioned to handle up to 5 damping values, 50 points per spectrum and 3000 ground acceleration values.

Availability

These programs are available at Earthquake Engineering Research Center, University of California, Berkeley. They will be installed on the CDC 7600 computer of Lawrence Berkeley Laboratory after purchase by the General Electric Company

Verification

SPECEQ/SPECUQ is recognized and widely used in industry with a sufficient history of successful applications to justify its validity per SRP 3.9.1, Section II.2.a.

Documentation of the programs is available in Reference 1.

Application

These programs will be used in computing design seismic response spectra for use in design of the piping systems.

Reference

Nigam, N. C., and Jennings, P. C., "Digital Calculation of Response Spectra from Strong Motion of Earthquake Response," California Institute of Technology, June, 1968.

A.84 SPHINX

The SPHINX code incorporates both one dimensional diffusion and transport theory in order to provide a standardized calculational scheme for generating multigroup cross sections which may be resonance self-shielded and space-energy collapsed to desired specifications. Interpolation schemes are used to compute shielding factors applicable to specific compositions. Collapsed group cross sections by reactor zone are calculated using flux-weighting.

Availability

The SPHINX code is currently available on the Nuclear Energy Systems CDC-7600 computer located at the Monroeville Nuclear Center.

Verification

The SPHINX code will become an integral part of the final design phase of the CRBRP project. Cross section data will be calculated for both the plate geometry critical experiments and the pin geometry CRBRP core. Reference design techniques, employing this cross section data, will first be applied to supporting critical experiments to establish the appropriate bias factors and uncertainties for critically, reactivity coefficients, control rod worths, reaction rates, etc. These bias factors and uncertainties will then be applied to the analysis of the CRBRP core in a consistent manner; the same reference design methods employing cross sections generated, in part, by SPHINX. This proposed verification scheme is similar to the current scheme which employs the ETOX, XSRES-1DX and ANISN codes to generate neutron cross sections.

Application

The basic input to SPHINX are cross sections and self-shielding factors in standard interface format as produced by the cross section processing code MINX. SPHINX will interpolate the self-shielding factors from MINX to the desired composition and temperature and use one-dimensional diffusion theory or one-dimensional transport theory to space-energy collapse the cross sections to the desired group structure. Cell homogenization of the composition and temperature corrected cross sections using transport fluxes is also available in SPHINX. The SPHINX code will ultimately replace the use of the XSRES-1DX and ANISN codes for nuclear analyses.

Reference

W. J. Davis, M. B. Yarbrough, A. B. Bortz, "SPHINX, A One Dimensional Diffusion and Transport Nuclear Cross Section Processing Code," WARD-XS-3045-17, August 1977.

A.85 SPRAY-3B

The SPRAY-3B Code (Ref. 1) performs the calculation of thermodynamics, heat transfer, mass transfer, and combustion of sodium droplets falling in a gravitational field. The droplets are confined to a given conical volume. The code predicts the pressure and the temperature transients of the gas in the cell as well as the temperature rise of cell wall structures.

Availability

The SPRAY-3B Code is available from General Electric in San Jose, California.

Verification

Comparisons of the code theoretical calculations with experimental data are provided in Reference 2 and 3. Validation of the SPRAY-3B Code was discussed with NRC on March 5, 1976. The summary of this meeting is presented in reference 4. Validation of the SPRAY-3B Code against additional test data is presently being done by the Project.

Application

The code SPRAY-3B is used for the analysis of sodium spray fires in gases having low (≤ 2 %) or high (21%) oxygen concentrations.

References

1) General Electric Specification #23A2842, "The SPRAY-3B Code".

2) P. R. Shire, "Reactor Sodium Coolant Hypothetical Spray Release for Containment Accidents Analysis: Comparison of Theory with Experiments," Proceedings of the Fast Reactor Safety Meeting, April 2-4, 1974. CONF-740401-P1, p. 473.

3) P. R. Shire, "Spray Code User's Report", HEDL-TME 76-94, 1977.

4) Letter, P. S. Van Nort to R. S. Boyd, "Summary of Meeting Held Between CRBRP Project and NRC to Discuss CACECO and Sodium Fires and SPRAY Codes used in CRBRP Analysis," March 10, 1976.

A.85A STALSS

The STALSS code analyzes the detailed steady state pressure drop distribution and three-way flow split in Secondary Control Assemblies. STALSS also calculates flow split accounting for the effect of uncertainties associated with the pressure drop measurements across the various major flow control devices and predictions of various resistances.

For a given set of boundary pressures at SCA inlet, upper outlet and lower outlet, STALSS calculates the flow split in the SCA through iterative process and detailed pressure drops. It also provides options to calculate probabilistic distribution of flow split and hydraulic scram-assist force at steady state.

AVAILABILITY:

The STALSS code is available on the Honeywell computer located at GE/NEBO - San Jose.

VERIFICATION:

Very good agreement was found in a preliminary code verification effort by comparing STALSS predictions of flow split with DYNALSS predictions. Validation against SCRS prototype test data is underway.

APPLICATION:

The STALSS code is used to predict detailed pressure drop distribution and flow split in SCAs. It also predicts the flow split uncertainties and hydraulic scram-assist force.

REFERENCE:

J. P. Wei, "Thermal and Hydraulic Analyses of CRBRP Secondary Control Rod System", CRBRP-GEFR-00542, January 1982.

A.86 SUPERPIPE (Proprietary Rockwell International)

SUPERPIPE is a finite element computer program designed to perform static load and stress analyses of linear and nonlinear elastic space frames and piping systems, and dynamic analyses of linear elastic systems of the same type.

Static loads considered include pressure, thermal expansion, dead-weight, point loads, specified displacements, static seismic loads, and fluid momentum loads.

Dynamic loads are calculated by the Response Spectra Method.

Special program features include non-linear elastic supports for static analyses, rigid components, and internal cold springing.

Availability

SUPERPIPE, developed originally for use on CRBRP, is only available to in house users at Atomics International, Canoga Park, California.

Verification

Verification of SUPERPIPE per SRP 3.9.1, Section II.2.b is documented in Reference 1. Comparisons were made with codes such as ADLPIPE, PIPDYN and SAP.

Application

SUPERPIPE will be for stress analysis for auxiliary system piping.

Reference

"Verification of SUPERPIPE", G. J. Madrid, Atomics International Report #N099TI530001, September 20, 1976.

A.87 TAP-A (& TAP-A-LARGE)

TAP-A is a digital computer program written in the FORTRAN language that solves steady state and transient heat transfer problems in multi-dimensional systems having arbitrary geometric configurations, boundary conditions, initial conditions, and physical properties. The program has the capability to consider the following modes of heat transfer and boundary conditions; internal conduction and radiation, free and forced convection, radiation at external surfaces, specified time dependent surface temperatures, and specified time dependent surface heat fluxes. The program will also handle space and time dependent thermal conductivity and heat capacity. In addition, the external boundary (environmental) temperatures may be functions of time.

Availability

TAP-A is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center. The version of the program used at ARD was generated in October 1975.

Verification

Results obtained using the TAP-A and TAP-A-LARGE computer codes have been shown to compare well with exact solutions and other analytic solutions available for a variety of problems and geometric configurations. One such problem is the case of one-dimensional steady state conduction heat transfer through a thick-walled cylinder; it is Benchmark Problem Number 4 of the Pressure Vessel and Piping 1972 Computer Programs Verification, edited by I.S. Tuba and W.B. Wright, ASME, 1972. Comparison of TAP-A to the Benchmark Problem is provided in Table A.87-1.

Table A.87-1 presents a comparison of the exact solution and that obtained with TAP-A. The agreement between the two solutions is excellent, with a maximum of 0.04% deviation between the TAP-A results and the exact solution.

Application

TAP-A is used to solve heat transfer problems that are not coupled with a hydraulic analysis.

TABLE A.87-1

COMPARISON OF TAP-A RESULTS

Radius(in.)	Temperature		TAP-A % Dev. of Theory
	Theory	TAP-A	
10.25	93.91	93.90	0.011
10.75	82.16	82.14	0.024
11.25	70.95	70.92	0.042
11.75	60.23	60.21	0.033
12.25	49.95	49.94	0.020
12.75	40.08	40.08	0.000
13.25	30.60	30.60	0.000
13.75	21.46	21.46	0.000
14.25	12.65	12.65	0.000
14.75	4.15	4.15	0.000

A.88 TAP-B

TAP-B is a digital computer program written in FORTRAN language to perform steady-state and transient analyses involving coupled fluid flow and heat conduction calculations for complex multi-dimensional flow systems separated by complex multi-dimensional heat-conducting solid systems. A maximum of two independent flow networks, or groupings, of flow channels can be handled by TAP-B. All modes of heat transfer and boundary conditions handled by TAP-A can be handled by TAP-B. Iterative finite difference techniques are employed to obtain problem solutions. TAP-B is capable of handling arbitrarily specified internal heat generation within both the solid and fluid models.

Availability

TAP-B is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center. The version used at ARD was developed by ARD as of April 1976. The code is not proprietary.

Verification

Verification will consist of comparing calculated results against closed form solutions, test data where available and applicable, and calculated results from other verified computer codes.

Application

TAP-B is used to calculate steady-state and transient flow distribution through complex flow systems and coupled effects between the fluid model and heat-conducting solid model, with consideration given to internal heat generation in both fluid and solid models.

A.89 TAP-4F (Proprietary Rockwell International)

TAP-4F is a descendant of the Lockheed Thermal Analyzer Program which has been in industry-wide use for many years and it is similar to TAP-A. It uses the finite difference method to calculate transient or steady-state temperatures in a network of thermal capacitances and conductors.

Availability

TAP-4F has been available at Rockwell International on the IBM 360 for several years. It is a Rockwell International proprietary computer code.

Verification

The computer program solutions to a series of benchmark problems with accepted results have been demonstrated to be substantially identical to those obtained by analytical solutions and/or numerical solutions obtained by Genreal Electric's HAP II and ASME's computer programs. This is documented in Reference (2). This constitutes verification per SRP Sections 3.9.1.II.2.b.

Application

TAP-4F is being used to calculate transient and steady-state temperatures in large networks of thermal capacitances and conductors. One such network represents a spent fuel assembly in a core component pot surrounded by the EVTM cold well.

References: (Rockwell International non-proprietary internal documentation)

1) "Thermal Analyzer Program Revision (TAP-4F)," E. Moody, Atomics International Report TI-099-411-015, August 1, 1974.

2) "Heat Transfer Code Verification," A. V. VonArx, Atomics International Report NO 36T1620-002, January 16, 1976.

A.89A TEMPEST

TEMPEST computes the steady state or transient temperatures and velocities in a three-dimensional flow field. The computer program solves the conservation equations of mass, momentum and energy for incompressible flows with small density variations, including the effects of turbulence. Either cylindrical or cartesian coordinate systems may be used. The flow field may include arbitrarily placed solids. The hydrodynamic solution is fully coupled with a heat diffusion solution in the solid materials. The flow model may consist of multiple flow regions separated by solid walls with different fluids in each region. A variety of initial and time dependent boundary conditions are allowed. The computer code features variable grid-spacing alogn any coordinate direction, arbitrary orientation of the coordinate system with respect to gravity, internal heat generation, the use of specified flow regions (input velocity and temperature), variable material properties for up to 50 material types, 20 time-dependent flow and temperature boundary tables, and specified or computed inflow and outflow boundaries.

Availability

The TEMPEST code is available from its developer, Battelle, Pacific Northwest Laboratory (BNW). Additionally, it has been installed on the Westinghouse Power System CDC 7600 computers located at the Monroeville Nuclear Center.

Verification

BNW has performed calculations and compared results with available experimental and analytical data for a range of applications, as described in Reference 1; additional verfication is planned and BNW will provide a verification document by the end of FY-82.

Application

TEMPEST is used primarily to calculate fluid steady state and transient temperature and velocities in plena such as the reactor outlet plenum; it has also been used to study flow stratification in pipes. It is useful for calculating natural circulation flows in isolated regions. The capability of TEMPEST to analyze three dimensional flow fields extends its utility beyond that available with the VARR-II hydrodynamic computer code.

Reference

1. D. S. Trent, M. J. Budden and L. L. Eyler, "TEMPEST: A Three-Dimensional Time-Dependent Computer Program for Hydrothermal Analysis", FATE-80-114, Battelle-Pacific Northwest Laboratory, January, 1981.

A.90 TFEATS (Westinghouse Proprietary)

TFEATS is a coupled two-dimensional temperature stress code and calculates, by finite element analysis, the temperature and stress fields for either axisymmetric or plane two-dimensional bodies with a variety of boundary conditions including prescribed displacements, loads and temperature boundary conditions.

Availability

TFEATS is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

Verification of the TFEATS computer program will be presented in the next interim CRBRP PHTS Piping Stress Report by showing correlation of benchmark problem results from other verified computer programs.

Application

This computer program is used to predict temperature distributions through the wall of piping components.

References

1. "TFEATS; A Computer Program for the Finite Element Thermal Stress Analysis of Plane or Axisymmetric Solids," Westinghouse Astronuclear Laboratory, Pittsburgh, PA, WANL-TME-1888, January 1969.

2. "CRBRP; Structural Evaluation Plan (SEP) for the HTS Piping System (Appendix F)," Westinghouse Advanced Reactors Division, Madison, PA, WARD-D-0087, June 1975.

A.91 TGRV

This program utilizes an electrical network analogy to obtain the temperature distribution of any given system as a function of time. The finite difference representation of the three-dimensional equations of heat transfer are repeatedly solved for small time increments and continually summed. Linear mathematics are used to solve the mesh network for every time interval. Included in the analysis are the three basic forms of heat transfer; conduction, radiation and convection, as well as internal heat generation.

Availability

This program is available on the IBM 370, model 165 computer of the Chicago Bridge and Iron Company.

Verification

Verification of approved programs is achieved through the use of sample problems which become a part of the permanent documentation folder. Verification is achieved through the use of one or more of the following:

1. Problems with a known closed form mathematical solution.

2. Comparisons with experimentally determined results.

3. Comparison with results obtained from other computer programs and reported in the literature.

4. Comparisons with results obtained from other approved CBI programs.

Application

TGRV will be used as a backup to E1606A for the temperature analysis of the head assembly and its components.

Reference

"CRBRP Design and Analysis of Head Assembly, Computer Programs and Program Control" Report 54500-4.2.3 Revision 0, Chicago Bridge and Iron, Oak Brook, Illinois, dated March, 1976.

A.92 THI-3D

The THI-3D code is a subchannel analysis code of the "rigorous" type, i.e., the flow distribution inside a gridded or wire wrapped assembly is calculated solving simultaneously the conservation equations of mass, momentum and energy. This subchannel analysis code expresses the thermal-hydraulic interaction in three dimensions. It solves the steady state non-linear boundary value problem with a pressure drop boundary condition. The wire wrap diversion cross flow, turbulent or grid enhanced cross flow between the channels are explicitly taken into account. Computed in this model are the temperature distribution of coolant, cladding, fuel and duct wall, and the size of the central void of the oxide fuel after thermal restructuring, including the effect of stainless steel swelling on thermal-hydraulic performance. The code has the capability to explicitly analyze primary control assemblies, accounting for bypass flow and coolant preheating prior to entering the absorber bundle. Both partial and the complete blockage problems can be analyzed. A development activity has been identified to include a transient capability in the code.

Availability

The THI-3D code is available through the Argonne National Laboratory Code Center at Argonne, Illinois. The program is used at Westinghouse on the Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

The THI-3D code results have been verified against experimental data from 19-rod, 91-rod and 217-rod assemblies. Documentation is included in the reference.

Application

The THI-3D computer code can be used to provide a benchmark solution to steady state flow and temperature distributions of gridded or wire wrapped assemblies. In CRBRP analyses, the code is used to calculate the coolant flow and temperature distribution inside the primary control assemblies.

Reference

1. W. T. Sha, R. C. Schmitt, "THI-3D: A Computer Program for the Steady State Thermal-Hydraulic Multi-Channel Analysis", ANL-8112, December 1975.

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A.93 THID/THTE (General Electric Proprietary)

This code computes transient and steady-state temperature solutions for three-dimensional heat transfer problems. The problem geometry is divided into nodal volumes. Temperature solutions are obtained by iterative solution of simultaneous algebraic equations for node temperatures derived from finite-difference analysis. The code can handle heat transfer by conduction, convection, and radiation. Material properties are included in temperature-dependent tables. The code is used to determine temperatures in various components.

Availability

THTD/THTE (May 1969/November 1971 version) is available for GE use at the Nuclear Energy System Division of the General Electric Company, San Jose, California on the Honeywell 6000 computer.

Verification

THID/THTE is recognized and widely used in industry with a sufficient history of successful applications to justify its validity, per SRP 3.9.1, Section II.2.a.

Application

It is used for the steady state and transient analysis of heat transfer problems in support of i.e. the design of the Secondary Control Rod System and the transient thermal analysis for SWRPRS operations.

Reference

F. C. Skirvin, "Users Manual for the THTD Computer Program (Transient Heat Transfer - Version D)", P.O. 036-926052-T0602, June 23, 1966.

A.94 TRANSWRAP

TRANSWRAP is a computer program to predict hydraulic transient phenomenon for complex piping systems subjected to sodium/water reactions. TRANSWRAP can calculate hydraulic transient pressure in circuits and determine the peak magnitudes of hydraulic pressure. It can also track transient interfaces of sodium-inert gas and interfaces of reaction product-sodium.

Availability

TRANSWRAP is available on the Honeywell 6000 computer at GE-FBRD. The code was provided by Atomics International.

Verification

The TRANSWRAP code utilizes the liquid hammer calculation methods of HYTRAN. A discussion of HYTRAN validation is presented in the PSAR Appendix A (A.51). Additional validation of the TRANSWRAP code will be provided by comparison with the Series I LLTR tests (See PSAR Section 1.5.1.).

Application

The TRANSWRAP code is applied to the Intermediate Heat Transport System (IHTS) and the Sodium Water Reaction Pressure Relief System (SWRPS) to provide a transient model of the evaporators, superheaters, pump, IHX, rupture discs, and associated piping. The primary objective of the code is to calculate pressure loadings on the steam generators and associated components in the IHTS and the SWRPRS due to sodium water reaction effects following the failure of one or more tubes in a steam generator module. The code also predicts velocities throughout the system, times at which rupture discs are actuated, the sodiuminert gas interface locations in the SWRPRS and the reaction products bubblesodium interfaces. The water injection rate is coupled to the reaction products bubble pressure which is in turn coupled to the sodium pressures at the bubblesodium interfaces. The one-dimensional method of characteristics is employed.

Reference

Atomics International Supporting Document Number T1-001-130-025, "TRANSWRAP - A Compressible Hydrodynamic Code for Large-Leak Sodium/Water Reaction Analysis", February 5, 1973.

A.95 TRIPLET-W or TRIPLET

TRIPLET-W is an ARD version of the TRIPLET code. TRIPLET-W solves the two-dimensional multigroup transport equation in planar geometries using a regular triangular mesh. Regular and adjoint, inhomogeneous and homogeneous (keff and eigenvalue searches) problems subject to vacuum, reflective or source boundary conditions are solved. General anisotropic scattering is allowed and anisotropic distributed sources are permitted. The discrete ordinates approximation is used for the angular variables. A finite element method in which the angular flux is assumed to be given by a low-order polynomial in each triangle is used to solve the discrete ordinates equations. Angular fluxes are allowed to be discontinuous across triangle boundaries, and the order of the polynomial is input data to the code. Both inner (within-group) and outer interation cycles are accelerated by either system or fine mesh rebalance. Sources, fluxes, S_n constants, and corss sections may be input from standard interface files. Creation of standard interface output files for S_n constants and scalar and angular fluxes is optional. All binary data transfers are localized in subroutines called REED and RITE. Flexible edit options, including dumps and restart capability, are provided. Implementation on the ARD computer system (CDC-7600) has required revisions in internal data manipulation to maximize the code's capability.

Availability

The TRIPLET-W code is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center. This version is derived from the Los Alamos Scientific Laboratory (LASL) version, released in October, 1973. Updates have been made at ARD to satisfy CRBRP shield design analysis requirements.

Verification

Code verification was performed by the originator of the code as indicated in the reference. Test cases transmitted to ARD from LASL as part of the code package and run on ARD's CDC-7600 system produce results consistent with LASL values. Agreement in results of the TRIPLET-W and DOTIIIW codes provide further verification.

Application

In the CRBRO application, TRIPLET-W is used to calculate detailed neutron and gamma flux distributions in various CRBRP components (e.g., primary and secondary control assemblies). Other applications are in calculations of detailed neutron flux distributions at component interfaces (e.g., at the interface between the hexagonal geometry removable radial shield and cylindrical geometry fixed radial shield). The detailed analysis results are used to define neutron fluence predictions for structural component material radiation damage studies.

Reference

W. H. Reed, T. R. Hill, F. W. Brinkley, and K. D. Lathrop, "TRIPLET: A Two-Dimensional, Multigroup, Triangular Mesh, Planar Geometry, Explicit Transport Code," LA-5428-MS, October, 1973.

A.96 TRITØN

The TRITØN code is a three-dimensional thermal-hydraulic code which calculates LMFBR core assemblies temperature accounting for inter-assembly heat transfer as well as wire wrap effects within the individual assemblies. Duct temperature calculations are the primary objective of TRITØN; however, assemblies mixed mean temperature and individual subchannel coolant and rod cladding temperatures are also calculated at the same time, thus providing a complete characterization of the assemblies thermal behavior properly accounting for inter-assembly heat transfer effects. TRITØN models any type of core assembly: fuel, inner and radial blanket, control and radial shield. A cluster of seven assemblies is analyzed in TRITØN simultaneously, thus, the temperature field in the central assembly of the cluster is properly influenced by inter-assembly heat transfer. By successively changing the central assembly, the entire core can thus be scanned. Sodium in the interstitial gap and heat generation by γ-heating in the control assembly ducts are considered in TRITØN. Duct temperature calculations can be performed for nominal conditions and accounting for uncertainty effects. Uncertainties are considered directly in the TRITØN calculations, thus providing an integral assessment automatically and simultaneously accounting for the effects of uncertainties and inter-assembly heat transfer. This procedure represents a noticeable improvement over the commonly used method of superimposing "a posteriori" the effect of uncertainties on calculated nominal temperatures. Combinations of positive (hot channel factors greater than unity) and negative uncertainties (less than unity) leading to the worst cross duct gradients can be analyzed.

Availability

TRITØN is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

The basic block of TRITØN, which is CØTEC, is being verified (see CØTEC). Partial verification of inter-assembly heat transfer calculations has been performed by comparing TRITØN with SUPERENERGY and CØRTAC/CØRTEM calculations. More detailed verification is planned as a future effort via code and data comparisons.

Application

TRITØN is used to solve the three-dimensional duct temperature distribution in the fuel, blanket, control and radial shield assemblies necessary for core restraint analyses. Mixed mean exit and rod cladding temperatures accounting for inter-assembly heat transfer are also calculated in supplement to NICER calculated temperatures under adiabatic assembly boundary conditions.

Reference

1. M. D. Carelli and C. W. Bach, "LMFBR Core Thermal-Hydraulic Analysis Accounting for Inter-Assembly Heat Transfer", Trans. Am. Nucl. Soc., 28, pp. 560-562 (June 1978).

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A.97 TRUMP

TRUMP solves a general non-linear parabolic partial differential equation describing flow in various kinds of potential fields, such as fields of temperature, pressure, and electricity and magnetism. Steady-state and transient in one, two, or three dimensions are considered in geometric configurations having simple or complex shapes and structures. Problem parameters may vary with spatial positions, time, or primary dependent values, such as temperature, pressure, or field strength.

Availability

TRUMP is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center. The code is not proprietary.

Verification

The validity of the TRUMP code has been established by the Lawrence Livermore Laboratory, by means of comparison of calculations with test data and exact, known solutions. Documentation concerning verification (in accord with SRP Sections 3.9.II.2.a, b and c) is included in the reference.

Application

TRUMP is used to solve coupled heat transfer and hydraulic problems where heat flux and flow fields are known functions.

Reference

A. L. Edwards, "TRUMP: A Computer Program for Transient and Steady-State Temperature Distributions in Multidimensional Systems," UCRL-14754, Rev. 3,. Lawrence Livermore Laboratory, University of California, Livermore, California, September, 1972.

A.98 VARR-II

VARR-II is a computer program which calculates steady state transient temperature and velocity fields in plena which have small density variations due to changes in fluid temperatures. The numerical technique solves the momentum, energy and continuity equations while including turbulent effects. The program allows for a variety of initial and boundary conditions as well as arbitrarily placed internal obstacles, and inlet and exit conditions. In addition, several methods of handling convective and diffusive fluxes are provided, permitting selection of the method most suited to the needs of particular problem configuration.

Availability

VARR-II is available on the Westinghouse Power System CDC-7600 computers located at the Monroeville Nuclear Center. The code has been submitted to the Argonne National Laboratory Code Center Library.

Verification

An experimental test program was initiated by Westinghouse ARD in 1974 to supplement analytical studies of hydro-dynamic problems with flow stratification in the Upper Plenum of the Clinch River Breeder Reactor. The validity of VARR-II has been demonstrated (in accord with SRP Section 3.9.1.II.2.c) by means of comparison of the analytical results with this experimental test data, and is reported in Reference 2.

Application

VARR-II is used in the two-dimensional analysis of hydro-dynamic problems having time dependent turbulent fluid flows with small density variations. The code is primarily used for calculation of steady state and transient conditions in the outlet plenum of the reactor. It allows for modeling a variety of boundary conditions, including obstacles, which represent internal boundaries, and inlet exit time varying flow and temperature conditions.

Reference

1. "VARR-II: A Computer Program for Calculating Time Dependent Turbulent Fluid Flows with Slight Density Variation", CRBRP-ARD-0106, Vols. 1, 2, 3, (Availability: USERDA Technical Information Center

2. Novendstern, E. H., Reese, J. C., Budden, M. J., "Prediction of the CRBRP Outlet Plenum Transient Response Following a Reactor Trip", ASME Paper 77-HT-30.

A.99 VENTURE

VENTURE is a code block designed to solve multi-neutron-energy-group, multi-dimensional neutronics problems. The finite-difference diffusion or a simple P_1 theory approximation to neutron transport is applied. The code solves usual neutronics eigenvalue, adjoint, fixed source, and criticality search (direct and indirect) problems, treating up to three geometric dimensions, maps power density and does first order perturbation analysis at the macroscopic cross section level.

Availability

VENTURE is currently available at the Oak Ridge National Laboratory (ORNL) computer facility. It is being made available at ARD on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

Additional verification of VENTURE will be performed by comparison of results with critical experiment measurements in the areas of power density distributions, control rod worths and other material worths.

Verification of one application of VENTURE has been performed by analysis of ZPPR-3 Phase 3 and Phase 3 Modified configurations. The intent of the ZPPR-3 series of critical experiments was to examine principal reactor parameters such as sodium void, power distributions and control rod worths in a typical Breeder Demonstration Plant. The analysis to date has verified both criticality calculations (see Table A.99-1 and sodium voiding reactivity worths (see Table A.99-2). The criticality bias factors ($\sim$1%) are similar to those determined with 2-D calculations. However, the sodium void worth uncertainties are significantly reduced when compared with 2-D RZ calculations of the same voiding patterns.

Application

VENTURE is used as a verification of two-dimensional calculations of material worths and power densities and to determine uncertainties for sodium void worth, power distributions, and control rod biases.

Reference

D.R. Vondy, T.B. Fowler, G.W. Cunningham, "VENTURE, A Code Block for Solving Multigroup Neutronics Problems Applying the Finite-Difference Diffusion-Theory Approximation to Neutron Transport," ORNL-5062, October 1975.

TABLE A.99-1

CRITICALITY CALCULATIONS FOR ZPPR-3 PHASE 3 AND MODIFIED PHASE 3 CORES

Configuration	Experimental $k_{effective}$	VENTURE Calculated $k_{effective}$	$\frac{C}{E}$
ZPPR-3 Phase 3	1.00031	0.9919	0.99
ZPPR-3 Modified Phase 3	1.00119	0.9905	0.99

TABLE A.99-2

SODIUM VOID WORTHS IN THE ZPPR-3 PHASE 3 MODIFIED CORE

Total No. Of Drawers Voided	Experimental Worth ($)	VENTURE Calculated Worth ($)[a,d]	
		Perturbation Theory	Eigenvalue Difference
42	0.4172	0.4225 (1.01)[c]	--
60	0.5672	0.5567 (0.98)	--
146	0.9968	1.0332 (1.04)	1.0299 (1.03)
228	1.3994	1.3808 (0.99)	--
510	2.0263[b]	2.0072 (0.99)	1.9753 (0.97)
632	1.7112	1.6896 (0.99)	1.5371 (0.90)

(a) $\beta = 0.003288$

(b) Maximum sodium voiding worth

(c) Calculation/Experiment

(d) Corrected to reflect worth of voiding in fuel only and not in control rods or channels.

A.100 WECAN (Westinghouse Proprietary)

WECAN is a proprietary Westinghouse computer program for the analysis of structures using the finite element method. The name is an acronym for Westinghouse Electric Computer Analysis. The program can be used to analyze efficiently both small and large finite element models of structures.

Availability

WECAN is available on the Westinghouse Power Systems CDC-7600 Computers located at the Monroeville Nuclear Center.

Verification

The WECAN program is in a continual state of development. The verification of the released version and demonstration of verification to NRC and other regulatory agencies is a continuing effort by Westinghouse. (Ref. 1,2 and 3)

Application

The WECAN computer program is used to solve a large variety of structural analysis problems. These problems can be one, two, or three dimensional in nature. It has the capability to do static elastic and inelastic analysis, steady state and transient heat conduction, steady state hydraulic analysis, standard and reduced modal analysis, harmonic response analysis, and both linear and linear transient dynamic analysis.

The program is based on the finite element method of analysis. The analyst must model, or idealize, the structure in terms of discrete elements and apply loadings and boundary conditions to these elements.

The stiffness (or conductivity) matrix for each element is assembled into a system of simultaneous linear equations for the entire structure. This set of equations is then solved by a variation of the Gaussian elimination method known as the wave front technique. This type of solution makes it possible to solve systems with a large number of degrees of freedom using a minimum amount of core storage.

The library of finite elements includes spars, beams, pipes, plane elements, axisymmetric solids of revolution elements, three dimensional solids, plates, plane and axisymmetric shells, three-dimensional shells, friction interface elements, springs, masses, dampers, head conduction elements, hydraulic conducting elements, convection and radiation elements.

WECAN is organized in such a way that additional structural elements can be added with a minimum of effort. Input formats are similar for all elements and all types of analysis. Input data used in the static analysis of a structure can be used for a dynamic analysis with only minor modifications. To complement the WECAN program, a system of pre and post-processors, called

To complement the WECAN program, a system of pre and post-processors, called WAPPP, is also available.

Reference

1. Westinghouse Electric Computer ANalysis Verification Manual, August 31, 1974, Westinghouse Research Laboratory, Beulah Road, Pittsburgh, Pa. 15235.

2. Westinghouse Electric Computer ANalysis Users's Manual, February 15, 1973, Westinghouse Research Laboratory, Beulah Road, Pittsburgh, Pa. 15235.

3. Westinghouse Electric Computer ANalysis Demonstration Manual, May 1, 1974, Westinghouse Research Laboratory, Beulah Road, Pittsburgh, Pa. 15235.

A.101 WESTDYN (Westinghouse Proprietary)

The digital computer program provides an elastic analysis of redundant piping systems subjected to thermal, static and dynamic loads. The system may contain a number of sections, a section being defined as a sequence of straight and/or curved members lying between two network points. A network point is (a) a junction of two or more pipes; (b) an anchor or any point at which motion is prescribed; or (c) a position of lumped mass. A network point may be free, or one or more if its six degrees of freedom may be constrained or displaced.

A member in the system may sustain prescribed loads or may be subject to elastic constraint in any of its six degrees of freedom. Also, at any location within the system members may be changed, masses concentrated, springs inserted, temperature conditions varied, materials and weld configurations changed, and body forces altered.

The code computes at each point within the piping system the forces, moments, translations, and rotations which result from the imposed anchor or junction loads, thermal gradients in the system, gravitational loads in any combination of the three orthogonal axes, wind loads, and earthquake disturbances. For seismic effects, a normal mode analysis is performed using three-dimensional response spectra. The resultant internal forces and moments are computed from the square root of the sum of the squares of the nodal forces and moments.

The code computes the stresses within the piping systems in accordance with USAS B31.1, "Power Piping" and Section III, "Nuclear Power Plant Components". The combination of load conditions to compute fatigue usage factors in accordance with Section III is included in this version.

Availability

The program is operational on the CDC-7600 computers of the Westinghouse Poqer Systems Computer Center.

Verification

Verification of WESTDYN, in compliance with SRP Sections 3.9.1.II.2a and b, is documented in reference WCAP-8252.

Application

WESTDYN will be usedin the flexibility analysis of the HTS sodium piping. The program is used to determine the stresses, displacements and flexibility loads at various locations along the piping. Loads on the deadweight hangers are also determined using WESTDYN.

References

1. "CRBRP: Structural Evaluation Plan (SEP) for the HTS Piping System (Appendix F)," Westinghouse Advanced Reactors Division, Madison, Pa., WARD-D-0087, June 1975.

2. "ADLPIPE: Static, Thermal, Dynamic Pipe Stress Analysis," Arthur D. Little, Inc., Cambridge, Mass., July 1973.

3. SCAP-8252, "Documentation of Selected Westinghouse Structural Analysis Computer Codes".

A.102 W-2DB (Westinghouse Proprietary)

W-2DB is a two-dimensional (XY,RZ,RΘ, triangular), multigroup diffusion code for use in fast reactor criticality and burnup analysis. The code can be used to compute k_{eff} and perform criticality searches on buckling, time absorption (α), reactor composition, and reactor dimensions by means of either a flux or an adjoint model. It can also calculate flux distributions for an arbitrary extraneous source and material burnup using a flexible material shuffling scheme.

Availability

The W-2DB code is currently available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

Verification of W-2DB results have been performed in many areas, specifically: power distributions, criticality and control rod worth. Numerous experiments from the Zero Power Plutonium Reactor (ZPPR) series of critical experiments have been analyzed with W-2DB to assess its accuracy. Calculations of the power distributions inthe ZPPR-3 and ZPPR-4 critical experiment program are described in subsection 4.3.2.2 of the PSAR. The root-mean-squared deviation between the calculated and measured fission rates is less than 2%. The range of criticality (keff) predictions, based on the analysis of ZPPR-2, ZPPR-3 and ZPPR-4 critical experiments with CRBRP reference design methods including W-2DB, is +0.3%Δk. This result is described in more detail in subsection 4.3.2.7.1 in the PSAR. Numerous experimentally measured control rod worths, in both ZPPR-3 and ZPPR-4, have been calculated with W-2DB. These analyses resulted in the establishment of an experiment-to-calculation bias factor of 1.03 and a root-mean-squared deviation between calculations and measurements of approximately 4% (1σ uncertainty). This work is fully described in subsection 4.3.2.6 of the PSAR.

Further verification of W-2DB will be performed by comparison of results with critical experiments measurements in the areas of power distributions, criticality, reaction rates, and control rod worths. A later Zero Power Plutonium Reactor (ZPPR) series of critical experiments in support of LMFBRs in general and the CRBRP in particular, will be analyzed with W-2DB to assess its accuracy.

Application

W-2DB, the basic two-dimensional, multigroup, diffusion code, is the heart of the reference design technique used in the nuclear analysis of the CRBRP. It is used to calculate criticality (keff), control rod

worths, power distributions, breeding ratio, fuel burnup and material inventories, and forward and adjoint fluxes. These parameters are employed in W-2DB and in other codes, such as PERT-V, to calculate the reactivity coefficients such as the Doppler temperature coefficient, sodium void reactivity worth, temperature and power coefficients, and radial and axial expansion coefficients. The detailed methods used to calculate all of these parameters are fully documented in section 4.3 of the PSAR.

Reference

W. W. Little, Jr., R. W. Hardie, "2-DB User's Manual-Revision 1," BNWL-831, Rev. 1, August 1969.

This reference documents the standard version of the 2-DB computer code; it does not contain the Westinghouse proprietary modification.

A.103 WRAPUP D

The WRAPUP D code analyzes the behavior of a single wire wrapped fuel rod during its life. The problem is defined by calculating the strain rate of the cladding and wire wrap as a function of axial position along the fuel rod. Wire and cladding stresses and strains as a function of time are calculated by solving first the elastic problem and then by integrating forward in small time intervals, recalculating the strain rate terms at each step. The wire wrap and cladding loads are calculated to maintain compatible fuel rod cladding and wire wrap deformation rates. The loads acting on the fuel rod which are included in calculating the cladding strains are:

1. Internal fission gas pressure
2. Axial loads on the fuel rod due to wire tension
3. Axial bending of the fuel rod due to wire wrap tension
4. Torsion of the rod due to wire tension
5. Internal fuel-cladding mechanical interaction pressure

In addition to the effects of these loads, cladding strains are also calculated due to thermal expansion, thermal creep, and irradiation creep and swelling. Wire loads are caused by the initial wire wrap tension and the differential expansion between the wire wrap and cladding. Wire wrap strains due to thermal expansion, thermal creep, and irradiation creep and swelling are also considered.

The results of the WRAPUP D code are utilized to decide whether:

a. The wire wrap can rupture due to overstressing and/or overstraining during service, or

b. The wire wrap can slacken, and slump away from the fuel rod to the extent that fretting wear is increased or direct contact between rods occurs.

Availability

The WRAPUP D code is available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

The analytical procedures in WRAPUP D have been verified by hand calculations and by comparing results with those of related codes, such as FRS. The wire performance predicted by WRAPUP D is currently being verified by comparison with the results from steady state and transient fuel rod irradiation tests.

Further data on the conditions which produce wire rupture and wire slackening will be obtained from future rod performance tests (see the response to NRC Question 241.15), and will be used to further verify the WRAPUP D wire performance predictions. WRAPUP D verification is discussed more completely in the Reference.

Application

The WRAPUP D code is used to calculate the wire wrap cladding interaction loads in the CRBRP fuel and radial blanket rods.

Reference

E. C. Schwegler, "Wire Wrap-Cladding Interaction in LMFBR Fuel Rods," Westinghouse Electric Corporation, Advanced Reactors Division, Madison, Pa., WARD-D-0149, July 1976.

A.104 XSRES-W1DX (Westinghouse Proprietary)

XSRES-W1DX is a multipurpose, one-dimensional (plane, cylinder, sphere) diffusion theory code for use in fast reactor analysis. The code is designed to compute resonance shielded and collapsed group cross sections. Resonance shielded cross sections are calculated using data (infinite dilution cross sections and resonance shielding factors) in the "Russian" format. Interpolation schemes are used to compute shielding factors applicable to specific compositions. Collapsed group cross sections by reactor zone are calculated using flux-weighting.

Availability

The XSRES-W1DX code is currently available on the Westinghouse Power Systems CDC-7600 computers located at the Monroeville Nuclear Center.

Verification

The XSRES-W1DX code is an integral part of the reference design method for calculating cross section data for both the plate geometry critical experiments and the pin geometry CRBRP core. Reference design techniques are applied to critical experiment measurements to determine the appropriate bias factors and uncertainties which are then applied to CRBRP calculations in a consistent manner. This analytical comparison with experimental results is fully documented in subsection 4.3.2 of the PSAR. Further verification will be performed based on EMC data. A set of test problems, specifically designed for code verification, will be developed.

Application

The primary application of the XSRES-W1DX code is the calculation of resonance self-shielded cross sections using "infinite dilute" cross sections from ETOX and resonance self-shielding factors for specific reactor compositions. In addition, criticality searches can be performed on material concentrations, region dimensions, and bucklings.

Reference

R. W. Hardie, W. W. Little, Jr., "1DX, A One-Dimensional Diffusion Code for Generating Effective Nuclear Cross Sections," BNWL-954, March 1969.

This reference documents the standard version of the 1DX computer code (XSRES is based on subroutines included in the 1DX code); it does not contain the Westinghouse Proprietary modifications.

A.105 MPHI Program

MPHI is a computer program that calculates moment-curvature (M-ϕ) relationships for reinforced concrete sections under thermal gradients and with an axial force P. The axial for P may be zero. The program has capabilities to account for non-linear material properties, variation of material properties with temperature, non-linear thermal gradients, tensile cracking and compressive crushing. The M-ϕ relationship provides information on the capacity of a section under a temperature gradient. The moment corresponding to zero curvature is also of interest because it represents the thermal moment for a section restrained against rotation.

For a given axial force P and temperature distribution, the moment-curvature relationship is determined using a numerical procedure that involves the following:

1. The section under consideration (Figure A.105-1), is divided into a number of elements by nodal points. The thermal strain ε_t is calculated at each node, i, as:

   $$(\varepsilon_t)_i = (T_i - T_{ref})\,\alpha(T_i)$$

   Where:

   T_i is the nodal temperature

   T_{ref} is the reference temperature

   $\alpha(T_i)$ is the average coefficient of thermal expansion at T_i.

2. A plane section is passed at a curvature ϕ with strains ε_a and ε_b at the two edges of the section. At node i the strain ε_i is:

   $$\varepsilon_i = \left(\frac{\varepsilon_a - \varepsilon_b}{H}\right) h_i + \varepsilon_b$$

   Where:

   H is the total depth of the section

   h_i is the coordinate of node i

   The mechanical strain at node i is:

   $$(\varepsilon_s)_i = \varepsilon_i - (\varepsilon_t)_i$$

3. Stresses, σ_i, are calculated at each node based on $(\varepsilon_s)_i$ and a stress-strain relationship which is defined in the input of the problem. Element forces are calculated based on the average of the stresses in the two nodes defining the element.

4. The axial force, P' and the moment, M, are calculated by summation of the element forces and their moments about the centroidal axis.

5. The value of P' is compared with the force under consideration (P) and if different the plane section is moved to a new position maintaining the same curvature ϕ and the process is repeated until convergence.

6. A new curvature is selected and steps 2 and 5 are repeated.

7. In this manner a ϕ versus M relationship is developed for a given P.

The program has options to develop the moment-curvature relationship for the following axial force, P, or restraint conditions.

a. Axial force specified in the input
b. No axial restraint ($P = 0$, displ. $\neq 0$)
c. Full axial restraint ($P \neq 0$, displ. $= 0$).

The last two options are special cases of the first and provide the capability to develop moment-curvature relationships for two extreme cases of axial restraint. The first option may be used for specific axial force values.

Tensile cracking of the concrete is accounted for automatically by the input σ-ε relations. Compressive crushing of a concrete element is assumed to occur in the part of the section where the strains exceed a limiting value which is input to the program. Degradation of a concrete element is assumed to occur in the part of the section where the temperature of the element exceeds a limiting value specified in the program. The part of the section where the strain or the temperature exceeds the limiting value is automatically removed from the section.

Availability

The MPHI program was developed by Burns and Roe and is available as a Burns and Roe in-house program in the Burns and Roe computer and in time-sharing CDC computers.

Verification

The program was verified by hand calculations. For this purpose a reinforced concrete section was considered under a temperature distribution and was divided into elements (Figure A.105-2). Material properties, for the purpose of this calculation, were assumed to be those shown in Figures A.105-3 to A.105-5. The M-ϕ relationships was developed for the case of full axial restraint and then, selected points on the relationship were calculated by hand. The moment-curvature points obtained by hand calculations are in complete agreement as shown in Table A.105-1. It should be pointed out that in addition to the values in Table A.105-1, the hand calculations provided a detailed check for the intermediate steps of the computer program such as strains and stresses to ensure that there are no errors that might affect the results under a different set of variables.

Further verification of the program was performed using the computer program ANSYS (Reference A.105-1). Limited analysis by ANSYS provided information for the moment corresponding to zero curvature (thermal moment) for the section in

Figure A.105-2 under full axial and not axial restraint. The model is shown in Figure A.105-6 and the properties in Figures A.105-3 to A.105-5. A comparison of the MPHI and ANSYS results is given in Table A.105-2.

Application

MPHI is used to calculate the moment-curvature relationship of reinforced concrete sections under temperature distribution and axial force. The moment capacity may be obtained from the M-ø relationship. In addition, the thermal moment of a section restrained against rotation may be obtained as that corresponding to zero curvature. (Figure A.105-7)

Reference

A.105-1 Computer Program ANSYS, Revision 3, Swanson Analysis Systems, Inc., Houston, Pennsylvania.

TABLE A.105-1

MPHI VERIFICATION RESULTS

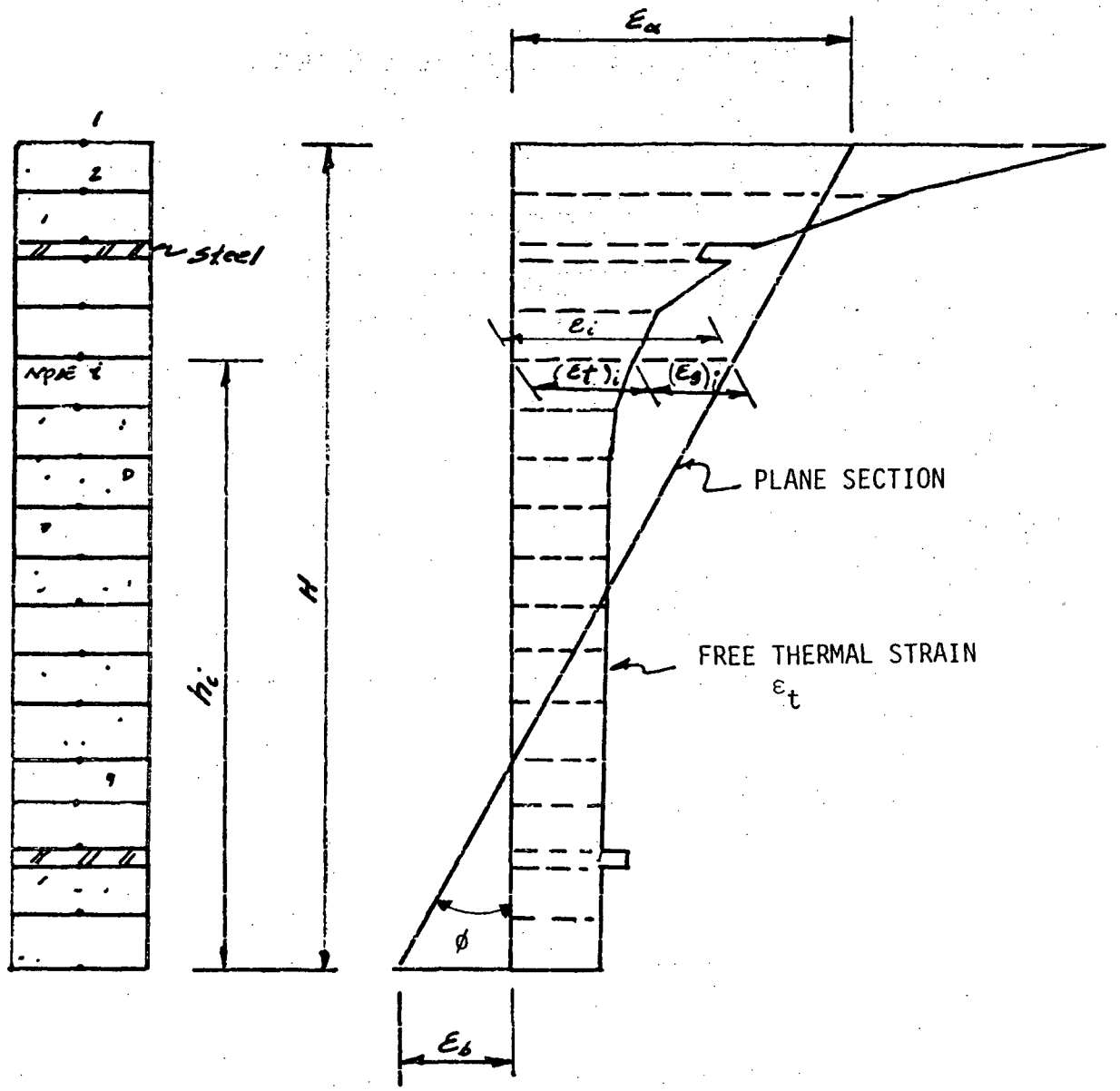
Curvature	Computer Results		Hand Calculations	
1/in.	Force lbs	Moment lbs-in	Force lbs	Moment lbs-in
0	-53,725	-27,327	-53,725	-27,327
+.00002	-53,892	+27,268	-53,892	+27,268
-.00002	-51,879	-76,738	-52,879	-76,738

Moment is positive when it creates tension on top (Node No. 1).
Negative axial force causes compression on the section.

TABLE A.105-2

COMPARISON OF RESULTS FROM MPHI AND ANSYS

CASE	MPHI RESULTS		ANSYS RESULTS	
	FORCE KIPS	MOMENT k-in	FORCE KIPS	MOMENT k-in
Curvature $\phi = 0$				
Full Axial Restraint	-53.7	- 27	-53.2	- 31
No Axial Restraint	0	-193	0	-188



MODEL OF CONCRETE SECTION

STRAIN DIAGRAM

FIGURE A105-1 TYPICAL MPHI MODEL AND STRAIN DIAGRAM

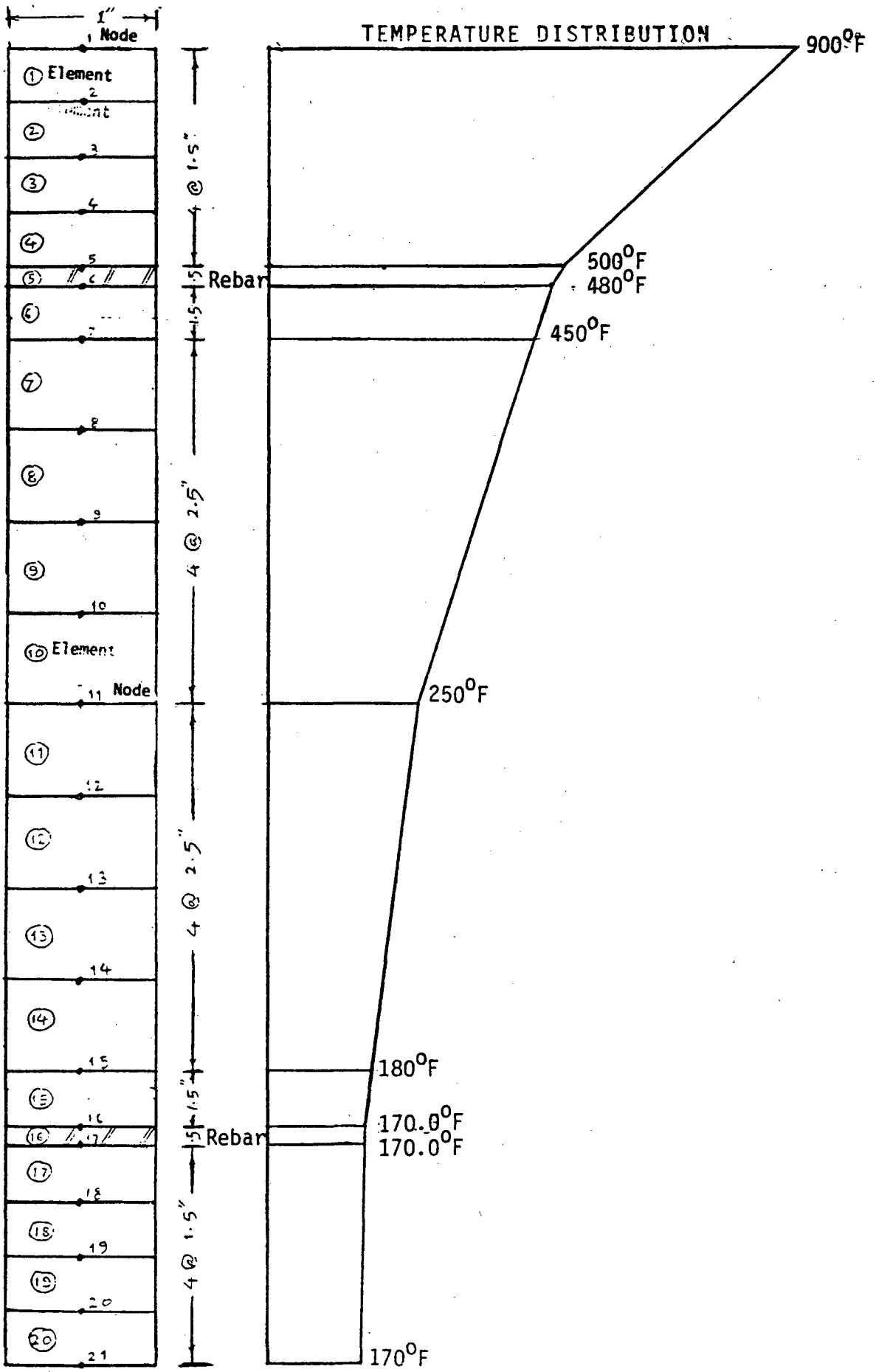


FIGURE A105-2 MODEL AND TEMPERATURES FOR VERIFICATION PROBLEM

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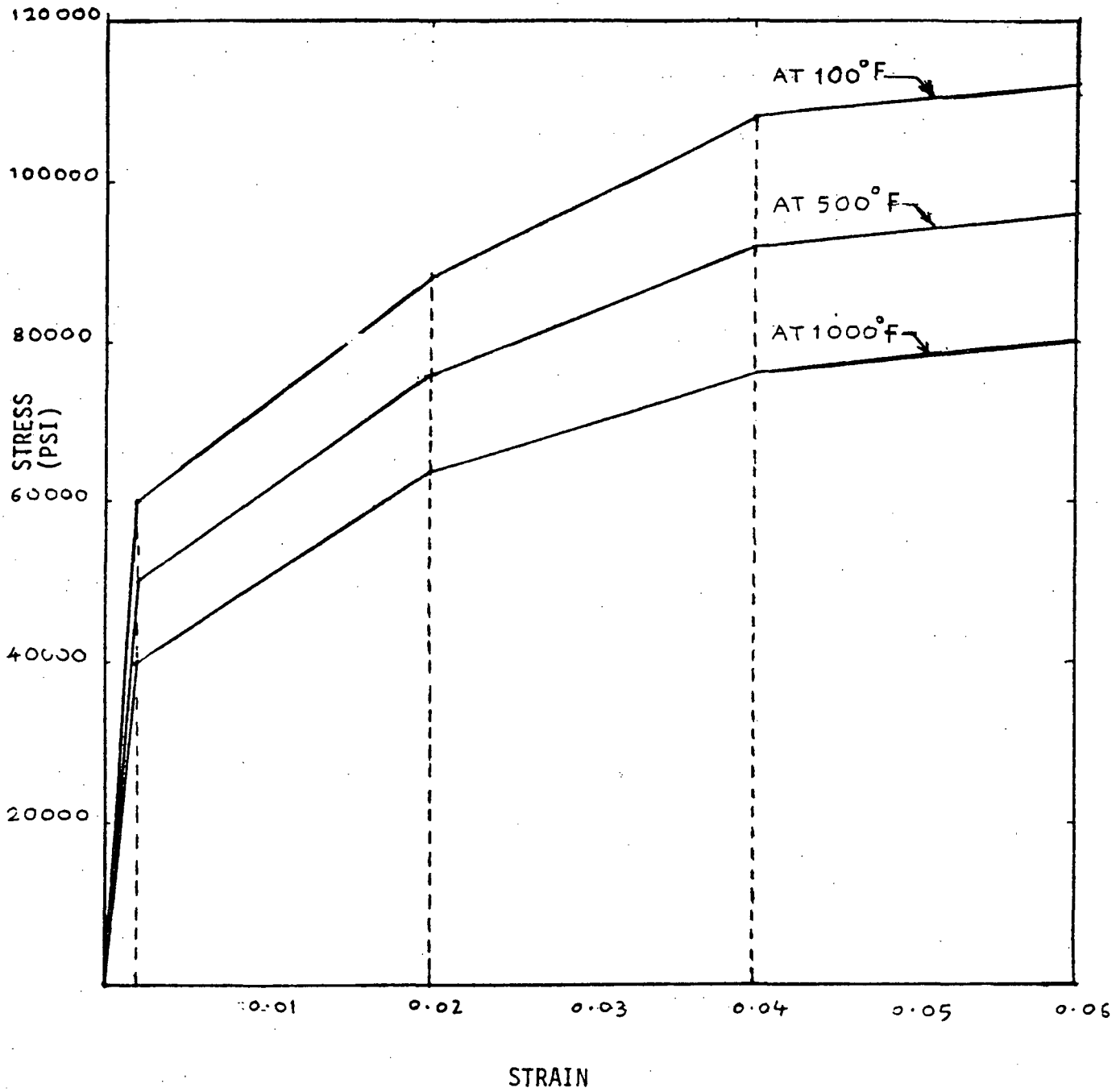
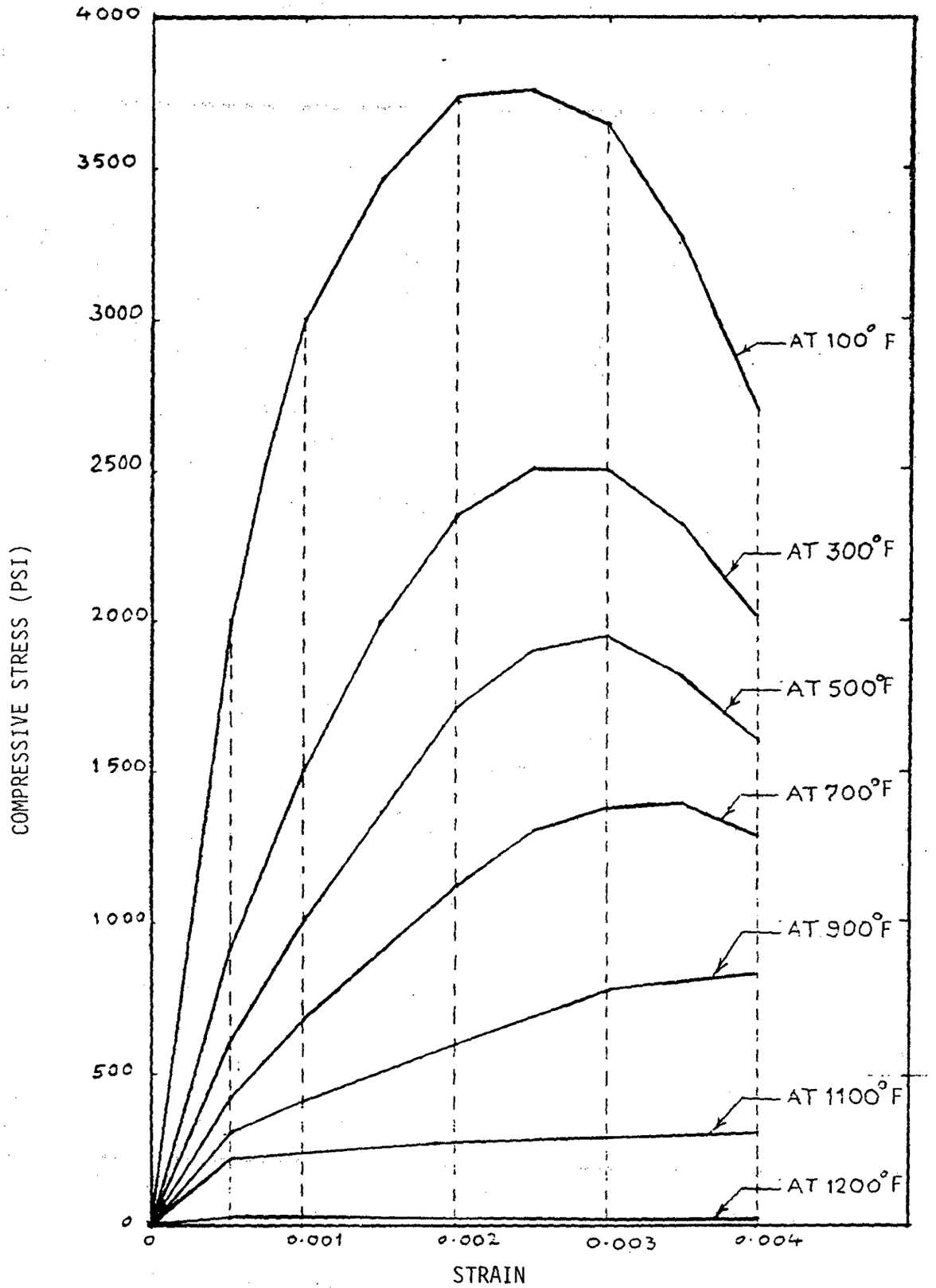


FIGURE A.105-3 STRESS-STRAIN CURVES FOR REINFORCING BARS - VERIFICATION PROBLEM



NOTE: Stresses in concrete are zero for tensile strains

FIGURE A.105-4 STRESS-STRAIN CURVES FOR CONCRETE - VERIFICATION PROBLEM

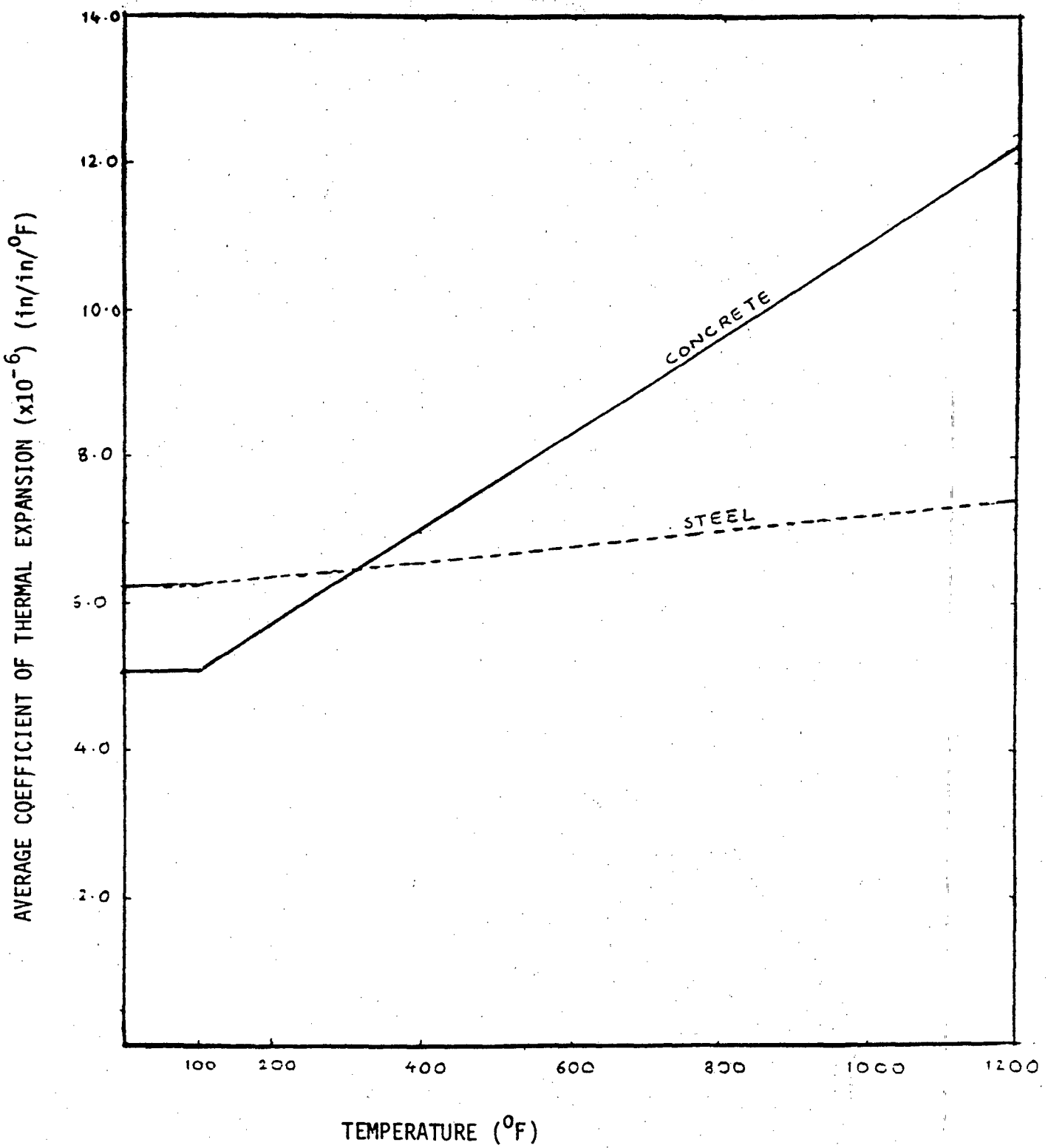


FIGURE A.105-5 COEFFICIENT OF THERMAL EXPANSION - VERIFICATION PROBLEM

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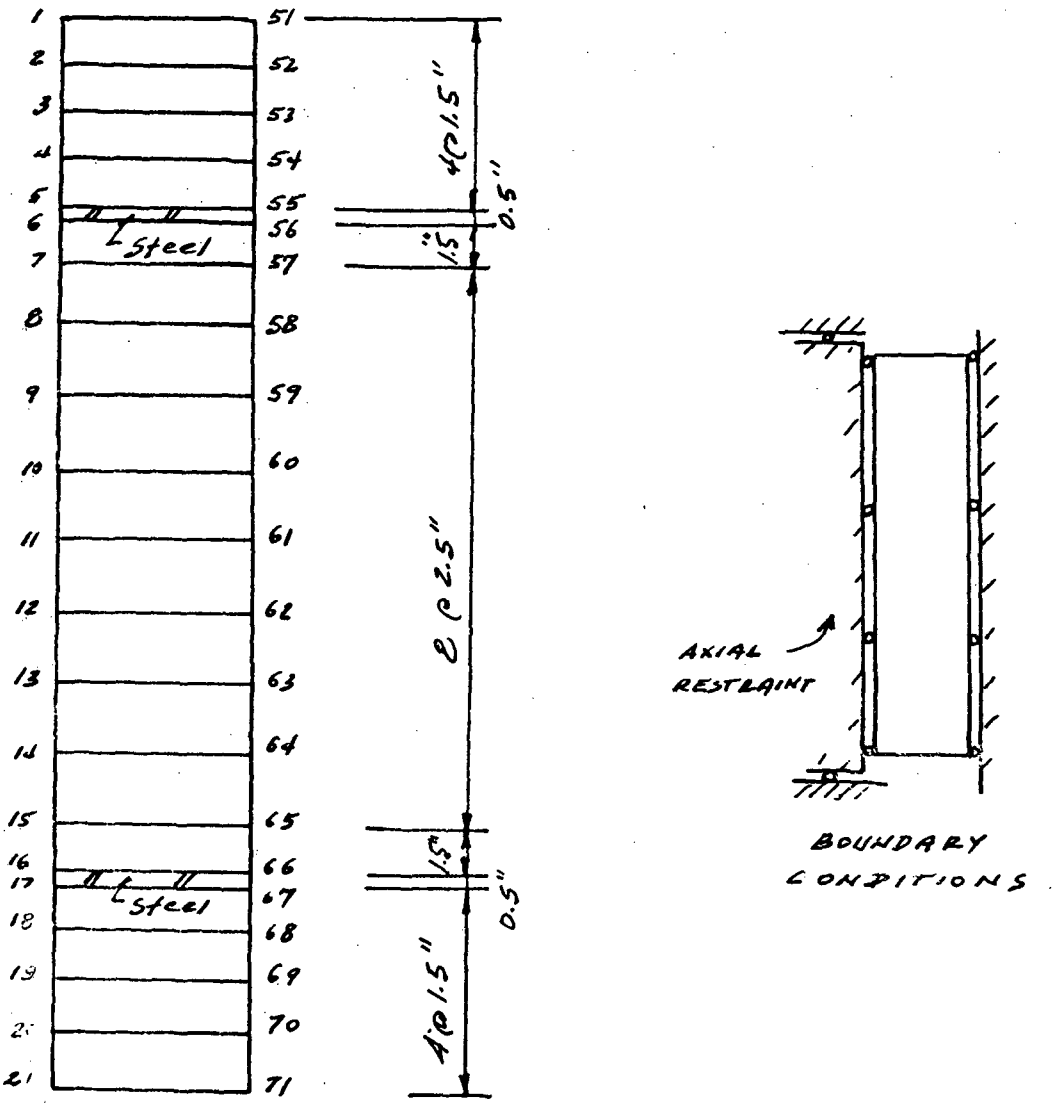


FIGURE A.105-6 ANSYS MODEL OF REINFORCED CONCRETE - $\phi = 0$

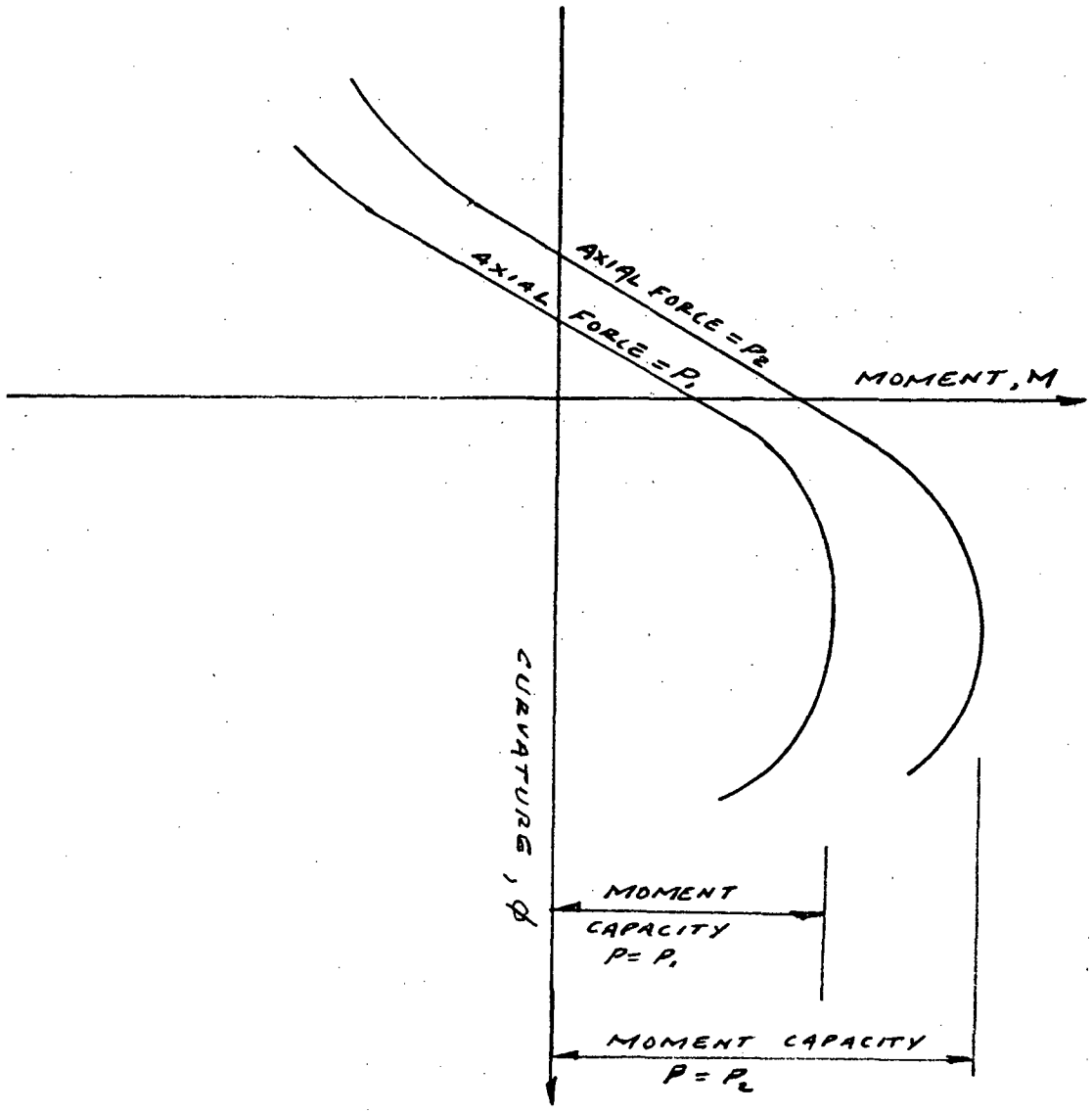


FIGURE A.105-7 TYPICAL MOMENT-CURVATURE DIAGRAMS

A.107 #722 (revised 9/78), Nozzle Reinforcing Area Check

This program calculates the nozzle reinforcing area available in shell, reinforcement, and nozzle wall and compares it to the area required by ASME Section VIII and/or Section III. Nozzle may have pressure or vacuum application.

Availability

This program (proprietary) is available at the CB&I office in Oak Brook, Illinois, on an IBM 3033N computer.

Verification

This program was verified by methods described and transmitted by Reference (1).

Application

The program is used to calculate the stress on the shell and the nozzle at the junction of the containment vessel.

References

(1) Letter HQ:S:82:169, J. R. Longenecker to P.S. Check, "Containment Vessel/Code Case(s) Analysis Working Meeting-Summary", dated December 9, 1982.

A.108 781, (revised 6/82) General Shell of Revolution

This program calculates the stresses and displacements in thin-walled elastic shells of revolution, when subjected to static edge, surface, and/or temperature loads with arbitrary distribution over the surface of the shell. The geometry of the shell must be symmetric, but the slope of the meridian is arbitrary. Since this program is based on classical shell theory, it has the same limitations. The shell thickness, physical properties of the materials, and loading may all vary arbitrarily around the circumference by using Fourier Series. There may be junctions or branches. The shape of the shell may be a cylinder, cone, sphere, torroid, ellipse, or parabola.

Availability

This program (proprietary) is available at the CB&I office in Oak Brook, Illinois, on an IBM 3033N computer.

Verification

This program was verified by methods described and transmitted by Reference (1), Section A.107.

Application

This program is used to calculate the stresses and displacements in the shell of the containment vessel.

References

None

A.109 #907, (revised 11/72) T/R Ratio and Geometry for Elliptical Head

This program calculates the T/R Radios and the geometry of an elliptical head for any major-to-minor axis ratio. The output includes the X and Y coordinates, the cone radius, and the radius of curvature at requested intervals along the head.

Availability

This program (proprietary) is available at the CB&I office in Oak Brook, Illinois, on an IBM 3033N computer.

Verification

This program was verified by methods described and transmitted by Reference (1), Section A.107.

Application

The program is used to calculate the X and Y coordinates, the cone radius and the radius of curvature for the ellipsoidal head of the containment vessel.

References

None

A.110 1017, (revised 9/78) Modal Analysis of Structures Using the Eigenvalue Technique

This program calculates the natural periods for a system consisting of lumped masses. The system can take into account anchor bolt stretch with appropriate modeling, foundation interation, and wave sloshing. The output includes stiffness and mass matrices as well as periods, deflections, shear, and moments for each mode. If special values are inputted, then maximum responses are obtained.

Availability

This program (proprietary) is available at the CB&I office in Oak Brook, Illinois, on an IBM 3033N computer.

Verification

This program was verified by methods described and transmitted by Reference (1), Section A.107.

Application

The program is used to calculate periods, forces, and moments in the shell of the containment vessel.

References

None

A.111 1027, (revised 1/82) Stress Intensities at Loaded Attachments; Cylinders and Spheres

This program computes the local shell stresses and stress intensities in the vicinity of an attachment to a spherical or cylindrical shell. Stresses are calculated for loads on the attachment given at the shell and centerline. The attachment is assumed to be rigid with solid cross-section. Formulas used in the program are based on work done by Professor P. P. Bijlaard as presented in Welding Research Council Bulletin 107 (WRC-107), August 1965 and revised March 1979.

Availability

This program (proprietary) is available at the CB&I office in Oak Brook, Illinois, on an IBM 3033N computer.

Verification

This program was verified by methods described and transmitted by Reference (1), Section A.107.

Application

The program is used to calculate local sheel stresses and stress intensities caused by external loads on the containment vessel.

References

Welding Research Council Bulletin #107.

A.112 1036, (revised 12/79) Maximum Stress Intensities in Jumbo Insert Plates

This program calculates the maximum stress intensity that is developed among several points of investigation around an externally loaded attachment on a jumbo insert plate in a cylindrical vessel. Stress analysis is in accordance with Welding Research Council Bulletin No. 107 (WRC-107).

Availability

This program (proprietary) is available at the CB&I office in Oak Brook, Illinois, on an IBM 3033N computer.

Verification

This program was verified by methods described and transmitted by Reference (1), Section A.107.

Application

The program is used to calculate shell stress intensities caused by external loads for the containment vessel.

References

Welding Research Council Bulletin No. 107.

A.113 1374, (revised 12/75) Shell of Revolution - Dynamics Program

This program determines the basic behavior of thin-walled elastic shells of revolution, when subjected to either static or dynamic pressures with arbitrary distribution over the surface of the shell. The geometry of the shell must be made up of spheres, torispheres, ellipsoids, and cylinders with stiffeners. Since the program is based on classical shell theory, it has the same limitations, namely:

1. The material is linearly elastic.

2. All displacements are very small.

3. Both the stress and strain normal to the surface of the shell are so small that they may be neglected entirely.

4. The shearing strains through the thickness are negligible so that normals to the midsurface remain straight and normal after deformation.

5. Both principal radii are much greater than the shell thickness so that

$$\frac{1}{R\phi + t} = \frac{1}{R\phi} \text{ and } \frac{1}{R\theta + t} = \frac{1}{R\theta}$$

Generally R/t must be greater than 10 for adequate accuracy.

Availability

This program (proprietary) is available at the CB&I office in Oak Brook, Illinois, on an IBM 3033N computer.

Verification

This program was verified by methods described and transmitted by Reference (1), Section A.107.

Application

The program is used to calculate the stresses and displacements in the shell of the containment vessel.

References

Same as A.108 781.

A.114 1392, (revised 3/79) Pipe Stresses Due to External Loads and Shell Stresses for Nuclear Containment Vessels

This program determines the stress intensity in a nozzle by analyzing the nozzle as a beam and adding the stress caused by internal or external pressure by superposition. Nomenclature and direction of loading is similar to that shown in WRC-107. The stresses are computed for four (4) points around the nozzle. These stresses are calculated near the shell (inside the area of shell reinforcement) and away from the shell (outside the are of shell reinforcement). The stresses found are then used to calculate contained stress intensities.

Availability

This program (proprietary) is available at the CB&I office in Oak Brook, Illinois, on an IBM 3033N computer.

Verification

This program was verified by methods described and transmitted by Reference (1), Section A.107.

Application

The program is used to calculate the stress intensity in penetrations of the containment vessel.

References

Not Applicable (uses fundamental physics equations only)

A.115 1671, (revised 9/78) Preprocessor for Programs 772, 1036, 1392

This program processes penetration loads and geometry, for a cylindrical containment vessel, to obtain specific loading combinations. Penetrations not on the cylinder must be analyzed separately. The output is in a format acceptable as input to programs 772, 1036, 1392.

Availability

This program (proprietary) is available at the CB&I office in Oak Brook, Illinois, on an IBM 3033N computer.

Verification

This program was verified by methods described and transmitted by Reference (1), Section A.107.

Application

The program is used to process output generated by CBI Programs 772, 1036, and 1392.

References

Not Applicable

A.116 1691, (revised 1/79) Three Dimensional Space Frame and Truss Analysis

The program analyzes two or three dimensional frames or trusses for member end forces, end moments, joint deflections and rotations. An analysis can be made on structures with rigid, hinged, or free support conditions, rigid or hinged member end conditions, and any number of loading conditions. Included in the program is a provision to use rectangular or cylindrical coordinates to describe the structure and a plotting option for a geometry check. The program can combine several loading conditions and can analyze the structure for member deadloads when the unit weight of the material has been input.

Availability

This program (proprietary) is available at the CB&I office in Oak Brook, Illinois, on an IBM 3033N computer.

Verification

This program was verified by methods described and transmitted by Reference (1), Section A.107.

Application

The program is used to calculate periods, forces, and moments, deflections, and stresses in 2 or 3 dimensional frames for the containment vessel.

References

STRESS, Program title developed at the Massachusetts Institute of Technology.

A.117 1823, (revised 10/79) Structural Analysis for Personnel Lock Bulkhead, Door, Hinge, and Latching Device

This program calculates stresses and some reactions for the personnel lock barrel, bulkhead, door, hinge, and latching device due to pressure and seismic loads. Loads and standard or modified coefficients for each stress summary sheet are inputted. Stresses or reactions are computed by multiplying the loads by the coefficients.

Availability

This program (proprietary) is available at the CB&I office in Oak Brook, Illinois, on an IBM 3033N computer.

Verification

This program was verified by methods described and transmitted by Reference (1), Section A.107.

Application

The program is used to calculate stresses and reactions in the air lock of the containment vessel.

References

Not Applicable (uses fundamental mathematical equations only)

A.118 BOSOR4

BOSOR4 is a computer program for stress, stability and vibration analysis of shells of revolution. The program was developed by D. Bushnell of Lockheed Missiles and Space Company (Reference 1).

The computer code is based upon the linear, elastic, thin shell theory. The structure should be axisymmetric. The program can handle various kinds of wall materials and loadings. Both mechanical and thermal loads are permitted in the analysis. In cases involving stress analysis of a shell for non-axisymmetric loading, the program finds the Fourier series for the loads, calculates the shell response in each harmonic to the load components with that harmonic, and superposes the results for all harmonics.

The program has an option by which the stability analysis of a shell can be treated as a bifurcation buckling problem and mathematically it is treated as an eigenvalue problem. The program also handles shell vibration as an eigenvalue problem and finds mode shapes and frequencies.

BOSOR4 uses a finite-difference scheme as a numerical technique in the solution of shell problem.

Availability

This program is available through CDC - Cybernet.

Verification

BOSOR4 is recognized and widely used in industry with a sufficient history of successful use to justify its validity.

Application

BOSOR4 has been used in the analysis of axisymmetrical structures.

Reference

(1) Bushnell D-Stress Stability and Vibration of Complex Branched Shells of Revolution: Analysis and User's Manual for BOSOR4. NASA/Langley Research Center, Hampton, Virginia. Contract NASI-10929.

A.119 CODES

CODES is a Burns and Roe computer program that designs reinforced concrete wall and shell structures per the requirements of ACI 318. The program calculates the reinforcement requirements due to axial load and bending, torsional moment, longitudinal and transverse shear. This is done for each of 14 load combinations and the maximum reinforcement areas for each group of elements are tabulated. In addition, the program contains an option wherein various loadings are combined and converted to principal forces. The components of the principal forces in the meridional and hoop directions are combined with the bending moments for the design of shell reinforcement.

Availability

CODES is developed by Burns and Roe.

Verification

The CODES verification has been done by hand calculation and is available in internal Burns and Roe documents.

Application

CODES is used in the design of reinforced concrete wall and shell structures.

A.120 EQUILIN

The computer program EQUILIN is designed to calculate an equivalent linear temperature distribution from a given non-linear distribution, according to the technique described in Reference (1).

The computer program is provided with information of a concrete section subjected to a non-linear temperature distribution. The program determines mean temperature of the section from the non-linear temperature. Also, an equivalent linear temperature distribution is found such that it produces the same uncracked moment about the centerline of the section as does the non-linear temperature distribution.

Availability

EQUILIN is developed by Burns and Roe and is available in the CYBER 176 Computer of CDC-CYBERNET System.

Verification

EQUILIN has been verified against hand calculations. A verification problem is attached and the results are found satisfactory.

Application

EQUILIN has been used to determine equivalent linear temperature distributions in thermal analysis of concrete structures.

Reference

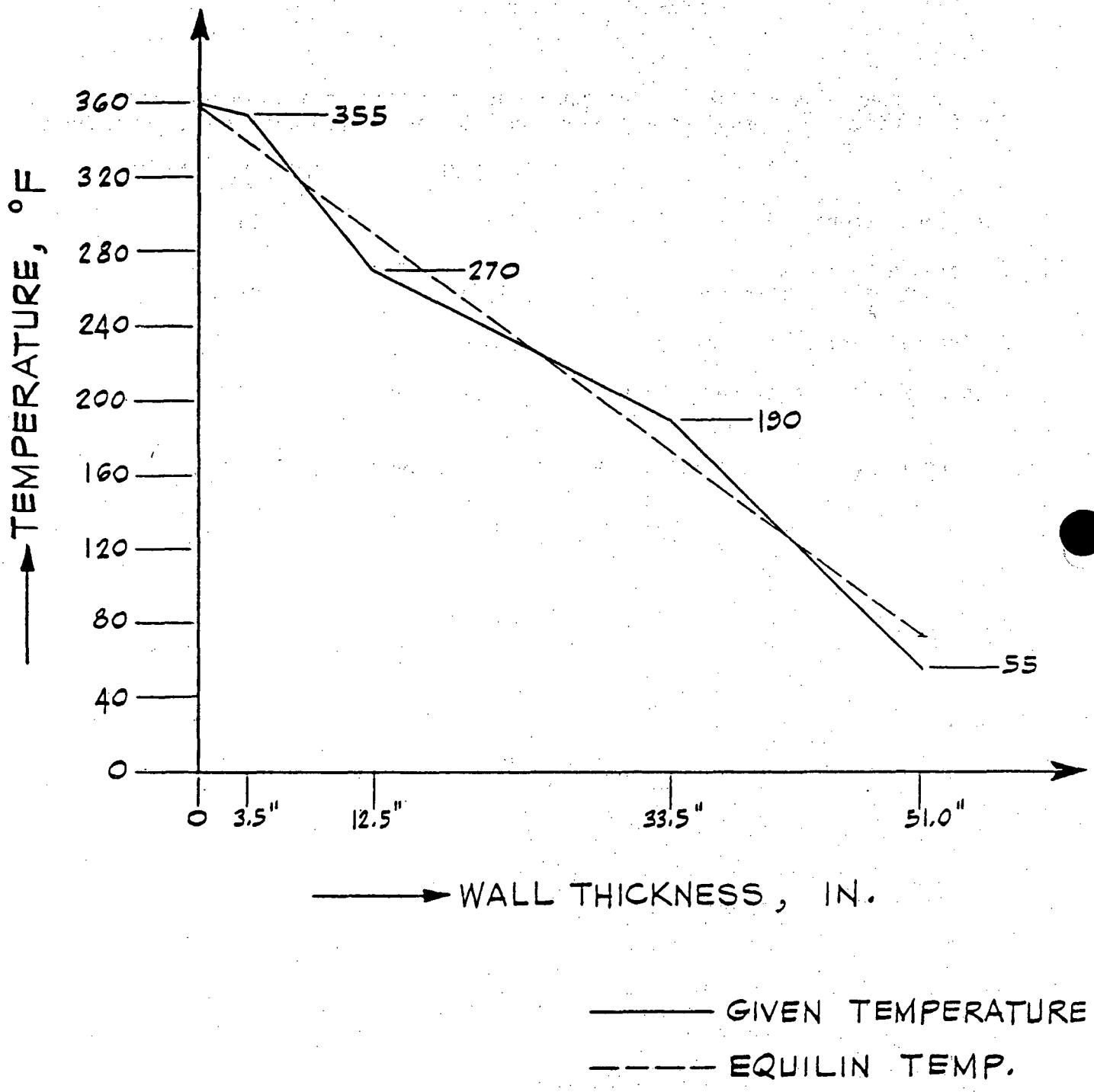
(1) ACI-349-76, Appendix A Commentary, Code Requirements for Nuclear Safety Related Concrete Structures

EQUILIN VERIFICATION

Attached figure shows a temperature profile through a concrete section. The following table provides a comparison between EQUILIN and hand calculations:

	EQUILIN	HAND CALCULATIONS
Mean Temperature	216.4	216.5
Difference in Temperature Through Wall Thickness	286.4	286.5
Inside Wall Temperature	359.6	359.7
Outside Wall Temperature	73.2	73.2

Equivalent linear temperature profile is shown dotted in Figure A.120-1.



EQUILIN RESULTS
FIGURE A.120-1

A.121 FLUSH

This is a computer program for approximate three dimensional analysis of soil-structure interaction problems.

The program was developed by J. Lysmer, T. Udaka, C. F. Tsai, and H. B. Seed of the University of California, Berkeley. The program is a further development of the complex response finite element program LUSH. FLUSH includes additional features such as transmitting boundaries, beam elements, an approximate three dimensional capabilities, deconvolution within the program, etc.

Availability

This program is available through CDC-Cybernet.

Verification

It is recognized and widely used in industry with a sufficient history of successful applications to justify its validity.

Application

This program has been used for soil-structure interaction in seismic analysis.

Reference

Lysmer, J.; Udaka, T.; Tsai, C.; See, H. B.; FLUSH, A Computer Program for Approximate 3-D Analysis of Soil-Structure Interaction Problems. Report No. EERC-75-30, November 1975, University of California, Berkeley, California.

A.122 MODPROP

This program calculates properties of a structure to be used in a "lumped mass" seismic analysis. The program calculates the masses and mass moments of inertia at specified elevations. It also calculates the member properties, such as area, shear area in two orthogonal axes (reference coordinates), area moment of inertia about these two axes, torsional rigidity, center of rigidity and centroid.

Availability

MODPROP has been produced by Burns and Roe and is available in the CYBER 176 computer of CDC - CYBERNET.

Verification

MODPROP has been verified against hand calculations. A verification problem is attached.

Application

MODPROP has been used to determine mass and stiffness properties of structures for seismic analysis.

MODPROP VERIFICATION

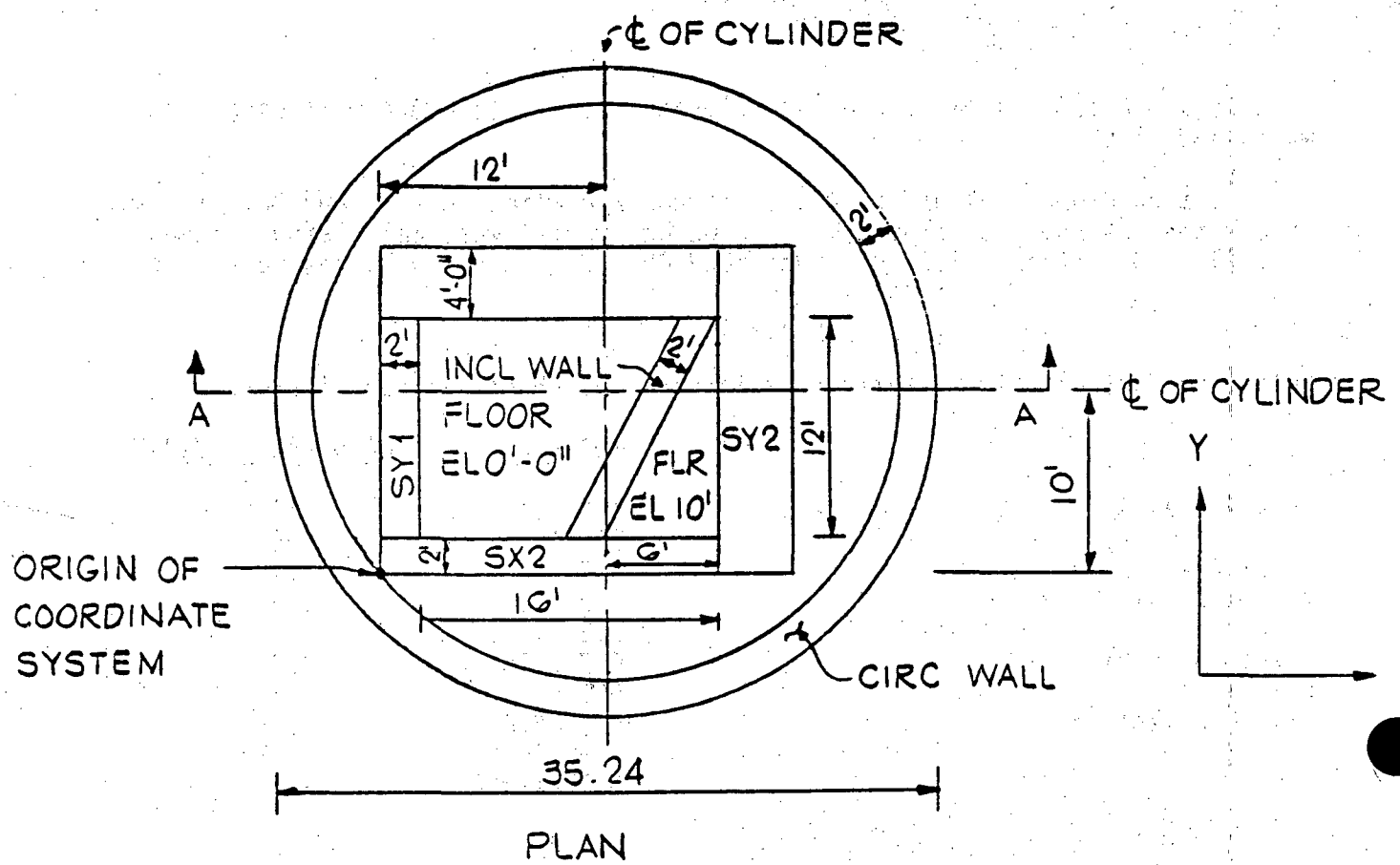
Figure A.122-1 shows a plan and elevation of a structure with concrete walls and slabs.

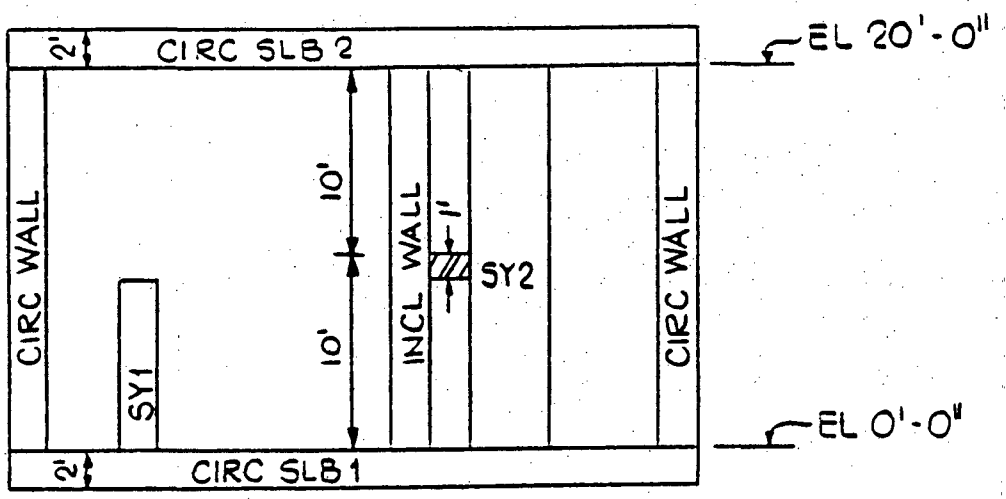
Section properties of the walls between Elevations 0.0 ft and 20.0 ft. were calculated. Also, mass properties (in weight units) were calculated between Elevation 10.0 ft and -2.0 ft. Both MODPROP and hand calculations were used.

Results

Units are feet and Kips. Moments of inertia are about centroidal axes.

	MODPROP	HAND CALCULATIONS
Shear Areas Ax =	224.43	224.4
Ay =	200.43	200.4
Total Area A =	415.68	415.6
Area Mass of Inertia Ixx =	0.37012E5	37012.0
Iyy =	0.37512E5	37512.0
Torsional Rigidity Kt =	0.7063E9	7.063×10^8
Weight, W =	953.9	953.9
Weight Moment of Inertia Iwx =	0.91827E5	91,829.0
Iwy =	0.96290E5	96,292.0
Iwz =	0.162192E6	162,195.0
Coordinates of Center of Rigidity, X =	14.53	14.53
Y =	10.36	10.36
Coordinates of Centroid: X =	12.74	12.735
Y =	9.96	9.957





SECTION A-A

MODPROP VERIFICATION PROBLEM
FIGURE A.122-1

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May 1983

A.123 RESPECTPLOT

This program computes response spectra from earthquake accelerograms digitized at equal time intervals. The generated response spectra represent the maximum responses of a damped single degree of freedom system. The program is based upon the techniques described in Reference (1).

Availability

RESPECTPLOT is a Burns and Roe modified version of SPECEQ/SPECUQ (Reference 1). It is available in the CYBER 176 Computer of CDC - Cybernet System.

Verification

RESPECTPLOT results were verified against STARDYNE Program of Reference (2) and the results are found satisfactory.

Application

RESPECTPLOT has been used to develop spectra from the time-histories.

References

(1) N. C. Nigam, P. C. Jennings, Digital Calculations of Response Spectra from Strong Motion Earthquake Forces, Earthquake Engineering Research Laboratory, California Institute of Technology, Pasadena, California, June 1968

(2) STARDYNE, Control Data Corporation, Publication No. 76079900

A.124 STRUDL

STRUDL is a comprehensive structural static and dynamic analysis and design program. It stands for STRUctural Design Language and conceived, developed and initially released by the Department of Civil Engineering, Massachusetts Institute of Technology, Cambridge, Mass.

Availability

The computer program is available through Georgia Institute of Technology, Atlanta, Georgia (GTSTRUDL) and McDonnell Douglas Automation Company (MCAUTO) of St. Louis, Missouri.

Verification

It is recognized and widely used in the industry with sufficient history of successful applications to justify its validity.

Application

It will be used for structural frame analysis of steel structures and supports.

References

(1) McDonnell Douglas Automation Company, "ICES STRUDL User Manual"

(2) Georgia Institute of Technology, "GTSTRUDL User Information Manual", Report No. SCEGIT-79-179, January 1979

A.125 THAVSA

This program calculates combined floor response spectra from spectra produced from independent seismic analyses for each of the earthquake directions (North-South, East-West, Veritical). It combines translational and rotational effects. The combination is based on equations 11, 12 and 13 of Appendix B of the Reference.

Availability

THAVSA was developed by Burns and Roe and is available in the CYBER 176 Computer of CDC - CYBERNET.

Verification

THAVSA was verified against hand calculations. The verification consisted of three steps:

1) Interpolation in a semilog spectrum plot (Figure A.125-1).

2) Verification of the calculations with equations 11, 12, 13 of Appendix B of WARD-D-0037. Results:

	THAVSA	Hand Calculations
Eq. 11	1.2630	1.2629
Eq. 12	1.3254	1.3254
Eq. 13	0.9084	0.9085

3) Plot Verification.

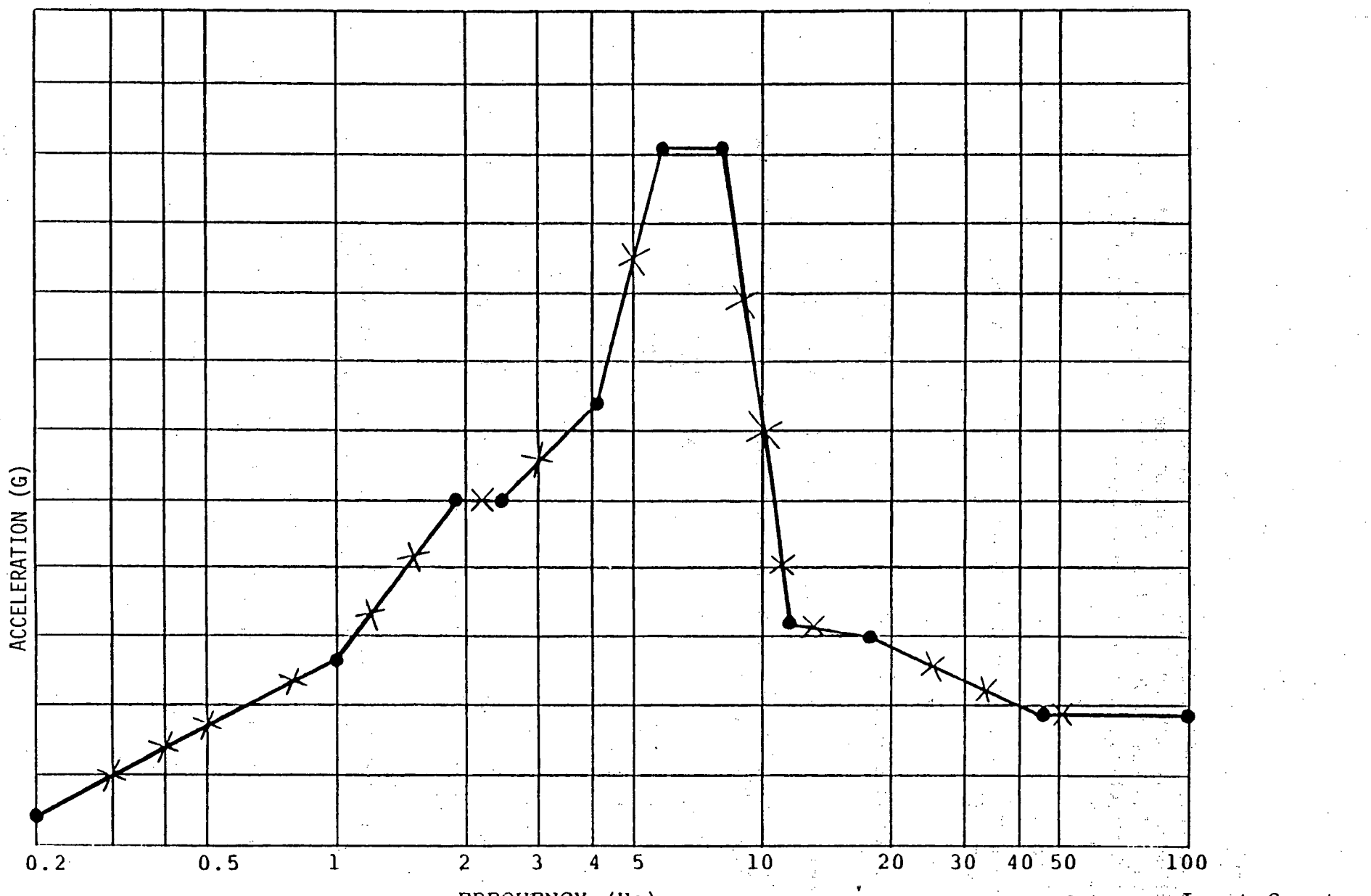
Figure A.125-2 shows a comparison of a spectral plot produced by THAVSA and the results of hand calculations.

Application

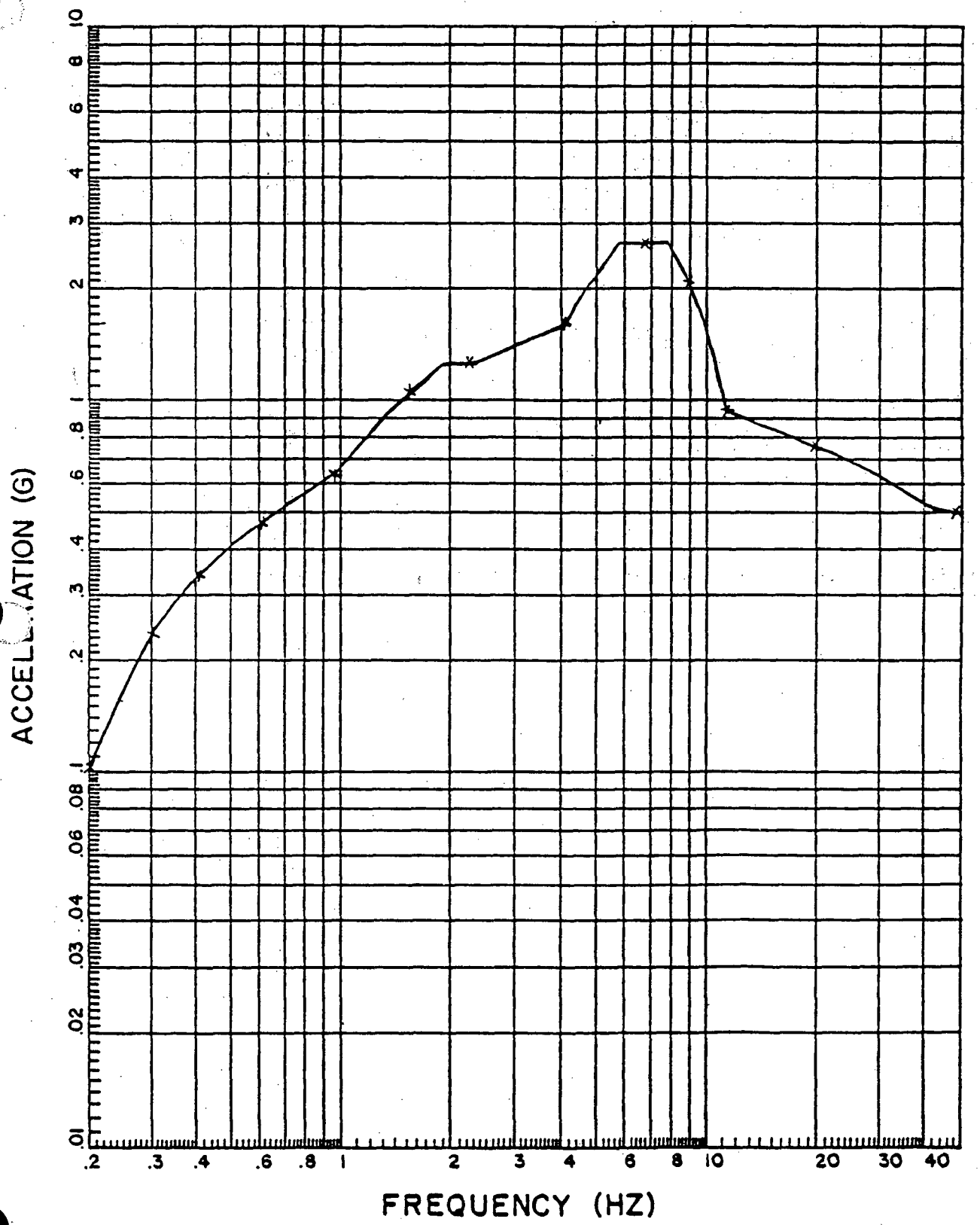
THAVSA has been used to produce Design Acceleration Response Spectra for Equipment Specifications and for the design of structural components.

Reference

WARD-D-0037 - CRBRP Seismic Design Criteria (PSAR Appendix 3.7-A).



INTERPOLATION VERIFICATION
FIGURE A.125-1



PLOT VERIFICATION
FIGURE A.125-2

——— THAVSA PLOT
X HAND CALCULATION

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May 1983

A.126 2DGENFRAME

2DGENFRAME is a time-shared computer program which performs the bending analysis of two-dimensional frames. Frame members can be rigidly attached or pin connected. Loading condition include combinations of forces at the joints, concentrated forces on the members. Distributed loads on the members, concentrated moments on the members. Distorsions of the members, and temperature changes in the members. Data can be entered interactively or from data files.

Availability

2DGENFRAME is commercially available to users through Control Data Corporation CDC KRONOS Time Sharing Computer Systems.

Verification

Verification of this program was done by hand calculation and is available in internal Burns and Roe documents.

Application

2DGENFRAME is used in the analysis of 2 dimensional frame structures.

CLINCH RIVER

BREEDER REACTOR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

APPENDIX B

GENERAL PLANT TRANSIENT DATA

PROJECT MANAGEMENT CORPORATION

APPENDIX B - GENERAL PLANT TRANSIENT DATA

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APPENDIX B GENERAL PLANT TRANSIENT DATA

B.1 CRBRP PLANT DESIGN DUTY CYCLE

This Appendix is a compilation of the events which comprise the CRBRP design duty cycle, together with an explanation of the method of selection of 'umbrella' transients. It should be noted that the inclusion of an item in this list signifies that it has been utilized for design purposes, but not that the event itself is necessarily regarded either as being credible, or to be expected as frequently as is indicated. Table B-1 presents the duty cycle frequency list. Table B-2 presents a preliminary listing of 'umbrella' events to be used as a basis for structural evaluation of the major heat transport system components.

B.1.1 Normal Events

B.1.1.1 N-1 Dry System Heatup and Cooldown, Sodium Fill and Drain

For design purposes, the heatup of the entire sodium system, exclusive of the steam generators, or of individual primary or intermediate loops will be treated as a temperature increase of the outer surface of the sodium containment from ambient (70°F) to 450°F at a constant rate of nominally
58 3°F/hr (desired rate is 25°F/hr),(10°F/hr for the reactor vessel). After a soak at 450°F surface temperature to preheat the internals to a nominal 400°F, the surface will be allowed to cool to 400°F. Similarly, cooldown will be considered as a decrease from 400°F to 70°F at a constant rate of 25°F/hr. (10°F/hr. for the reactor vessel). Plant systems will be filled with argon at one atmosphere. Each heatup cycle will be preceded by three cycles of pressure reduction to as close to full vacuum as practical and back filling with argon to one atmosphere. Each heat up cycle will be followed by one pressure cycle from ambient to maximum attainable vacuum with back filling to one atmosphere using argon. It is assumed that all sodium containing piping and components will be heated by electrical heaters mounted external to the piping, component, or guard vessel, as applicable. The steam generator modules and the steam-water system will be heated from the water side using an auxiliary heat source. Following the heatup, the primary and intermediate systems are filled with 400°F sodium. The systems are drained and backfilled with argon at one atmosphere prior to cooldown below 400°F.

B.1.1.2 N-2 Normal Startup

B.1.1.2.1 N-2a Startup from Refueling

59 The plant startup event from refueling is a heatup transient between the normal refueling temperature of 400°F and the temperature conditions that exist at a minimum operating power level of 40% thermal power. For design purposes, the primary system sodium temperature will increase essentially isothermally at an average rate of 50°F per hour between 400°F and 600°F. This heatup rate will be achieved by utilizing the sodium

47 pumps at 100% flow and may include a minimal amount of reactor power. Between 600°F and the system conditions existing at 40% thermal power, the hot leg temperature will change at an average rate of 150°F per hour. This change in temperature will be accomplished by making discrete steps in power level which will result in temperature changes at a rate of 1°F per second for 25 seconds every 10 minutes. The primary cold leg and intermediate system hot and cold leg temperatures will vary between 400°F and their appropriate temperatures for the 40% thermal power level. Primary sodium flow rate during this portion of the heatup is taken to be constant at 40% of full flow with appropriate intermediate flowrates. The water/steam side of the steam generators will follow the nuclear island temperatures from 400°F to the operating temperature level of 600°F with water/steam circulation through the evaporator module and steam through the superheater. The water/steam pressure will be varied from about 425 psig to ∿1500 psig as required to meet the operation conditions at 40% thermal load. Steam flow will vary as required to heat the turbine and reject the excess heat generated by the reactor.

B.1.1.2.2 N-2b Startup from Hot Standby

59 The plant startup event from hot standby is a heatup transient between the hot standby temperature of 600°F and the temperature conditions which exist at a minimum operating power level of 40% thermal
47 power. This event is the same as the second part of N-2a (starting at 600°F).

B.1.1.3 N-3 Normal Shutdown

B.1.1.3.1 N-3a Shutdown to Refueling

59 The plant shutdown event to refueling is a cooldown transient between the temperature conditions which exist at a minimum operating power level of 40% thermal power to the normal refueling temperature of 400°F. This event is assumed to be essentially N-2a reversed in time with the exception that the reactor will be taken subcritical when the primary hot leg temperature has been reduced to slightly above 600°F and the sodium pumps will be run at pony motor speed during the cooldown between 600°F and 400°F. The cooldown between 600°F and 400°F is handled by the Protected Air Cooled Condensers (PACC) with assistance from the main feedwater and turbine bypass system to accelerate the cooldown. When refueling conditions are reached the PACC's handle the entire heat load and the main steam stop and feedwater isolation valves are shut. The main feedwater system is used intermittently after this to provide makeup as required for long term
47 leakage.

B.1.1.3.2 N-3b Shutdown to Hot Standby

59 The plant shutdown event to hot standby is a cooldown transient between the temperature conditions which exist at a minimum operating power level of 40% thermal power to the hot standby temperature of 600°F. This event is assumed to be N-2B reversed in time, with the reactor taken subcritical when the primary hot leg temperature has been reduced to slightly above 600°F. When the protected air cooled condenser can handle the decay
47 heat load, the main steam stop and feedwater isolation valves are shut.

B.1.1.4 N-4 Load Following

B.1.1.4.1 N-4a Loading and Unloading

The plant design loading and unloading events are conservatively
58 represented by a continuous and uniform ramp load change through the range of
47 40% to 100% of full load. This load range is the maximum permissible consistent with the reactor control system, which is designed to accommodate automatic load following capability while maintaining rated steam conditions.
47 Rate of load change is up to 3.0% per minute. Load changes in this region are accomplished by linearly varying primary and intermediate sodium flows with power while holding turbine inlet pressure constant.

B.1.1.4.2 N-4b Load Fluctuations

In addition to normal plant loading and unloading (N-4a), there will be load fluctuations resulting from changing electrical network demands. For design purposes, these events are conservatively represented by continuous
47 and uniform ramp load changes through the range of 80% to 100% of full load. This load swing is used since it results in the largest temperature variation in the system for the given 20% load variation. Rate of load change is up to 3.0% per minute. As in loading and unloading, load changes are accomplished
47 by linearly varying primary and intermediate sodium flows with power while holding turbine inlet pressure constant. For calculation of the system conditions during a load fluctuation, it should be assumed that equilibrium conditions are reached between the ramp power changes.

B.1.1.5 N-5 Step Load Changes of ±10% of Full Load

This event involves step changes in generator load equivalent to ±10% of full generator load within the load range of 40% to 100% of full
47 load. These events are assumed to be occasioned by normal disturbances in the electrical network into which the plant output is tied. The nuclear island is assumed to respond to the load change by changing flow and power at the rate of 3% of rated power per minute.

B.1.1.6 N-6 Steady State Temperature Fluctuations

This event consists of the sodium temperature variations produced by power and flow fluctuations within the control system deadband. This fluctuation is taken to be $\pm 6^{o}F$ peak to peak for 30×10^{6} cycles and is based on expected deadband fluctuations. The fluctuations may arise from the deadband of the power loop of the control system (±2%) which would result in a temperature fluctuation period of about 24 seconds at full flow. Since
58 the system is not expected to limit cycle, the frequency is considered to be conservative.

B.1.1.7 N-7 Steady State Flow Induced Vibrations

This event consists of the vibrations in the sodium heat transport system produced by the fluctuations in sodium pressure due to the interaction between the vanes in the impellers and the pump volute. Steam generator system water side fluctuations caused by other system vibrations and frequency are determined on a system basis. 58 47

B.1.2 Upset Events

All upset events are terminated at hot shutdown unless otherwise specified. 58 59

B.1.2.1 U-1 Reactor Trip

This transient includes real scrams due to malfunctions (including rapid reactivity transients) which cause a PPS trip level to be exceeded and spurious scrams covering those situations in which a PPS trip level is not actually exceeded; but a scram occurs due to a fault in the PPS, control system or plant instrumentation.

B.1.2.1.1 U-1a Reactor Trip From Full Power with Normal Decay Heat

This transient involves a trip of the reactor (release of primary and secondary control rods) followed in 300 msec by the tripping of the main sodium pumps. The sodium pumps coast to pony motor speed, which results in a sodium flow of about 10% of full flow in both the primary and intermediate loops. Feedwater and recirculation pumps remain energized and feedwater flow is controlled by the normal control system that responds to drum level and feedwater/steam flow signals. The turbine is tripped on a low throttle pressure signal. The turbine bypass valve starts opening coincident with the turbine trip. The initial decay heat level for this transient is normal 100% decay heat. The Protected Air Cooled Condenser for each loop is turned on within 240 seconds of the reactor trip. 47 58 47 58 47 47

B.1.2.1.2 U-1b Reactor Trip from Full Power with Minimum Decay Heat

The same operational sequence of event U-1a is assumed for this transient. The minimum decay heat level is assumed for an initial condition.

B.1.2.1.3 U-1c Reactor Trip from Partial Power with Minimum Decay Heat

The same operational sequence is assumed for this transient as for reactor trip from full power. Initial power level for this transient is 40% and the initial decay heat is minimum decay heat. Since larger initial thermal differences can occur in the steam system at part power than full power, this transient is included as a separate case. 58

B.1.2.2 U-2 Uncontrolled Control Rod Movement

This is a general category of events, which result from control system malfunctions. The U-2 category includes six events: an uncontrolled rod

insertion from full ΔT initial conditions, an excessive startup step power change, and four rod withdrawal cases. These events are identified in the duty cycle for the purpose of providing assurance of their consideration in the overall transient analysis task and as a basis for the determination of plant protection system requirements.

B.1.2.2.1 U-2a Uncontrolled Rod Insertion

A single rod is inserted at a rate which causes a 1/2%/second reduction in thermal power due to an assumed malfunction of the controller on that rod. (This event is not to be confused with a rod drop, which is an unlatching of the rod resulting in a free fall of the control rod.) The sodium, feedwater, steam and recirculation flows are held constant, as is the turbine admission valve inlet pressure. It is assumed that this event occurs when full system ΔT's are present. The power level at the beginning of the transient is 100%. An operator manually trips the plant four minutes after event initiation.

B.1.2.2.2 U-2b Uncontrolled Rod Withdrawal From 100% Power

An uncontrolled withdrawal of one control rod is assumed to cause
47 the reactor power to increase at 0.5% nuclear power per second from 100% to 115% (just below the high flux trip point). A manual reactor trip is assumed after 5 minutes. Sodium flows are maintained at initial values until the trip occurs and the feedwater and steam flows are increased appropriately for the increase in reactor power. For the core, it provides the worst case
47 sustained overpower design event.

B.1.2.2.3 U-2c Uncontrolled Rod Withdrawal from Startup with Automatic Trip

The initial conditions for this event are hot standby with minimum decay heat. Primary and intermediate main pump motors are started and sodium flows are increased to 40 percent. Uncontrolled withdrawal of one control
47 rod at 0.5% nuclear power/second then occurs. During the withdrawal, all sodium flows remain at initial values. A reactor trip is initiated by a flux-delayed flux subsystem.

8.1.2.2.4 U-2d Uncontrolled Rod Withdrawal from Startup to Trip Point With Delayed Manual Trip

The initial conditions for this event are hot standby with minimum decay heat. Primary and intermediate main pump motors are started and sodium flows are increased to 40 percent. Uncontrolled withdrawal of one control
47 rod at 0.5% nuclear power/second then occurs. The power ramp is terminated just before the flux-delayed flux trip point is reached. After 10 minutes, the event is terminated by a manual scram. Flows are maintained constant at initial value.

B.1.2.2.5 U-2e Plant Loading at Maximum Rod Withdrawal Rate

From initial plant conditions of 40% reactor thermal power, 40% sodium flow, and ∿35% electrical output, the station supervisory control causes the plant to load at the nominal rod withdrawal speed. Sodium flows and reactor power increase from 40% to 100% at 1%/second. The turbine increases output at the same rate. The drum level control operates normally to approximately match feedwater flow to steam flow. The feedwater pump control increases feedwater pump speed at 1% per second. No scram occurs.

B.1.2.2.6 U-2f Reactor Startup with an Excessive Step Power Change

The reactor startup with an excessive step power change is a normal startup as defined in event N-2a. In addition it is assumed that during the course of the startup, there is a power change resulting in temperature changes at a rate of 1°F/second for 50 seconds followed by a constant outlet temperature for 1150 seconds.

B.1.2.3 U-3 Complete or Partial Loss of One Primary Pump

There are two events in this category: partial loss of primary flow in one loop and the loss of power to one primary pump main motor.

B.1.2.3.1 U-3a Partial Loss of Primary Pump

The primary flow in one loop is assumed to decrease from 100% to
47 a level immediately above the flow ratio trip setting (at approximately 70% flow) due to a ramp down in pump speed at minimum coastdown rate. The primary sodium flows in the two unaffected loops as well as the intermediate flows in all three loops remain at their initial values. No action is taken to terminate the event for 10 minutes. A manual trip terminates the event at that point. This transient provides an envelope to encompass control malfunction and operator errors causing mismatches in the primary to intermediate flow ratio at design values. The transient will result in an increased reactor outlet temperature and a redistribution of temperatures within the IHX in the affected loop.

B.1.2.3.2 U-3b Loss of Power to One Primary Pump

The primary pump in one loop is assumed to coast down to pony motor speed. The other primary pumps are assumed to be under speed control. The intermediate pump speeds in all loops remain at initial values until reactor/pump trip. A reactor trip is initiated when the ratio of normalized primary to intermediate pump speed is less than 0.7. Following
47 the reactor trip, the remainder of the pumps and the steam/water side are treated as for the normal scram.

The transient provides an envelope that encompasses events which would cause the pump to be tripped or which results from control failures more severe than those in U-3a or significant operator errors in controlling 47 primary loop flow.

B.1.2.4 U-4 Complete or Partial Loss of One Intermediate Pump

There are two events in this category: a partial loss of intermediate flow in one loop, and the coastdown of one intermediate pump to pony motor speed.

B.1.2.4.1 U-4a Intermediate Pump Control Failure

The intermediate flow in one loop is assumed to ramp down at 1%/ 47 second from 100% flow to 70% flow (a level immediately above the speed ratio trip setting). All other flows remain controlled at their initial values. No action is taken to terminate the event for ten minutes. At that point, the incident is terminated by a manual scram. The event is characterized by an increase in intermediate hot leg and primary cold leg temperature and a decrease in the intermediate cold leg temperature.

B.1.2.4.2 U-4b Loss of Power to One Intermediate Pump

The intermediate pump in one loop is assumed to coastdown to pony motor speed. The other primary and intermediate pump speeds are assumed to remain at initial values until the reactor/pump trip. A reactor trip is initiated when the normalized primary to intermediate pump speed ratio exceeds approximately 1.43. Following the trip, the remainder of the pumps and the steam/water side are treated as for the normal scram.

B.1.2.5 U-5 Reduction or Loss of Feedwater Flow

B.1.2.5.1 U-5a Loss of AC Power to One Feedwater Pump Motor

58 47 The steam plant includes three 50% capacity feedwater pumps. Two pumps operate at any one time. The third (spare) main feedwater 47 pump will be started automatically upon loss of one operating feed pump. Automatic startup of the spare feedwater pump avoids a significant transient on any component. Failure of the spare feedwater pump to automatically start will cause a rapid reduction in feedwater flow rate to approximately 60 percent of full flow. This will cause the feedwater flow control valve in each loop to go full open, and the temperatures of the water entering each of the evaporators will increase. Although the loss of AC power to one feedwater pump motor should result in an automatic start of the standby (spare) feedwater pump, for conservatism in design, it is assumed that the spare does not start and the plant is tripped on low drum level one minute after the initiation of the events. The one minute trip delay is based on a drum inventory equivalent to a 30 second holdup at full flow. 47

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B.1.2.5.2 U-5b Loss of Feedwater Flow to All Steam Generators

This transient includes three cases: (a) loss of one feedwater pump with failure of its outlet check valve; (b) loss of feed pump suction, and (c) closure of all feedwater throttle or feedwater isolation valves. These events all result in a decrease in drum level due to feedwater loss. A reactor trip will be initiated on steam-feedwater flow ratio. SGAHRS will be actuated automatically in all three loops on a low drum level signal to
47 prevent loss of all plant cooling. This event is similar to event U-18 with the exception that the recirculation pumps remain in operation.

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B.1.2.7 U-7 Sodium Pump Speed Increases

B.1.2.7.1 U-7a Primary Pump Speed Increase

The event is assumed to involve a control system malfunction that demands 100% flow from an initial condition of 40% flow and 40% reactor power.
47 All three primary pumps increase their speeds at a rate of 1%/second. A trip is initiated by the primary to intermediate speed ratio subsystem. The event results in a down ramp of core outlet temperature. This event provides envelope coverage for speed increase of a single pump under the same conditions as well as for startup of a main pump motor from pony motor speed.

B.1.2.7.2 U-7b Intermediate Pump Speed Increase

This event assumes an increase in all three intermediate pump speeds from an initial condition of 40% flow and 40% reactor power at a rate of 1%/
47 second. A trip is initiated by the primary to intermediate speed ratio subsystem. This event provides envelope coverage for speed increase of a single intermediate pump under the same conditions as well as for startup of a main pump motor from pony motor speed.

B.1.2.8 U-8 Primary Pump Pony Motor Failure

Following a normal plant trip, the pony motor on one of the three primary pumps fails to engage. The affected pump coasts down and stops. Pony motor pumping action in the other two loops results in flow reversal in the loop while intermediate sodium flow remains constant. The check valve limits reverse flow to 1100 gpm. The event is characterized by an abnormal temperature distribution in the IHX, a rapid decrease in primary hot leg temperature in the affected loop, and increasing core temperatures
47 due to the bypassed flow.

B.1.2.9 U-9 Intermediate Pump Pony Motor Failure

Following a normal plant trip, the pony motor on one of the three intermediate pumps fails to engage. The affected pump coasts down and stops. The transient is characterized by an up transient in the cold leg of the primary circuit of the affected loop, and a long term mismatch in primary to intermediate system flows in the affected loop since the intermediate system flow in the affected loop will be provided by the natural circulation driving head only.

B.1.2.10 U-10 Inadvertent Closure of Steam Generator Steam/Water Module Isolation Valve

This event includes isolation valve closures for a single evaporator or superheater module.

B.1.2.10.1 U-10a Evaporator Module Inlet Isolation Valve Closure

The normally open module inlet isolation valve is assumed to instantaneously close. The other evaporator module experiences an increase in recirculation flow. The affected evaporator module will experience increasing exit quality, and the evaporator module and its outlet line to the steam drum will boil dry. The affected evaporator module will then experience a large up transient.

Reactor trip will be initiated from a high evaporator module outlet sodium temperature signal.

B.1.2.10.2 U-10b Superheater Module Inlet Isolation Valve Closure

The normally open inlet isolation valve is assumed to instantaneously close. The loss of superheater steam flow will result in a reactor trip, initiated from a steam-feedwater flow ratio subsystem. The continued addition of heat to the water in the evaporators of the affected loop will cause the drum and evaporator pressures to increase, and the affected loop drum and evaporator exit pressure relief valves will open. The feedwater control valve, which may begin to close in response to the initial feedwater/steam flow mismatch, will open to maintain the drum level, balancing the

steam flow through the drum and evaporator power relief valves. Steam will be vented until the SGAHRS can remove the heat. The superheater module, as well as the sodium cross-over piping and evaporator sodium inlets, experience severe up temperature transients.

B.1.2.10.3 U-10d Superheater Module Outlet Isolation Valve Closure

The normally open outlet isolation valve is assumed to instantaneously close. The superheater steam flow in the affected loop will be stopped, resulting in a pressure increase in the affected loop steam generating components. Turbine steam flow will accordingly be reduced. Superheater steam flow will be reestablished through the superheater exit power relief valves. A reactor trip will be initiated by a steam feedwater flow ratio subsystem.

B.1.2.11 U-11 Inadvertent Isolation and Blowdown of Steam Generator

The events are assumed to be initiated by one of the following: a) inadvertent operator activation, b) inadvertent activation caused by
47 a component failure, or c) operator response to a water to sodium leak indication. This transient results in the water isolation and dumping of either a single evaporator module, the superheater or both evaporators and the superheater in an individual loop. For design purposes, an intermediate loop sodium drain is assumed. The events in this group terminate at re-
47 fueling conditions.

B.1.2.11.1 U-11a Water Side Isolation and Dump of Both Evaporators and the Superheater

This event is initiated by a signal which is assumed to instantaneously close the normally open isolation valves in the inlet water lines to the evaporators and in the inlet and outlet steam lines of the superheater
47 of a steam generator. The dump valve in the water side inlet to each evaporator and the power relief valves in the evaporator and superheater water/steam side outlets are assumed to simultaneously open. Pressure from the steam drum closes the check valve in each evaporator outlet line. These valves serve to provide outlet line isolation of the evaporators from the
47 steam drum and avoid blowing down the drum. The steam/water side pressure decreases until the power relief and dump valves shut. The modules are then pressurized on the water/steam sides with nitrogen at 200 psig. A reactor trip is assumed to occur 4 seconds after the isolation valves
58 shut due to steam flow/feed flow mismatch. The event is characterized by up transients of the steam generator modules and intermediate sodium pump in the affected loop. The unaffected loops see transients similar to a reactor trip from full power.

B.1.2.11.2 U-11b Water Side Isolation and Dump of an Evaporator Module

This event is initiated by a signal which is assumed to instantaneously close the normally open isolation valve in the evaporator inlet recirculation line. Simultaneously, the dump valve in the water side inlet line of the affected evaporator and the power relief valves on the evaporator exit line are assumed to open. Pressure from the steam drum closes the check valve in the evaporator outlet line. This valve serves to provide outlet
47 line isolation of the evaporator and avoids blowing down the steam drum. The steam/water side pressure decreases until the power relief and dump valves shut. The evaporator water/steam side pressure is then maintained

at 200 psig using the nitrogen supply. The plant is tripped manually by
47| the operator or on high evaporator module outlet sodium header temperature. For the unaffected loops, the event is similar to a reactor trip from full power. Decay heat removal is maintained through the unaffected steam generator or modules.

B.1.2.11.3 U-11c Steam Side Isolation and Dump of Superheater

This event is initiated by a signal which is assumed to instantaneously close the isolation valves in the steam inlet and outlet lines to the superheater and to open the relief valve in the steam header at the superheater module outlet. This event interrupts steam flow from one of the three steam generator systems and a reactor trip results due to steam-feed flow mismatch. Heat transferred from the sodium to the water side in the evaporators is released by opening relief valves at the evaporator module exits and on the steam drum. The steam side pressure is maintained at 200 psig using the nitrogen supply.

47| B.1.2.12 U-12 Loss of Feedwater Flow to One Steam Generator Loop

This event is specified as an inadvertent closure of the feedwater inlet isolation valve or the feedwater throttle valve which controls the feed-
47| water flow to the steam drum of one of the three steam generator loops. Closure of the valve is assumed to occur in 0.3 seconds. This event is characterized by an increase in the water inlet temperature to the evaporators and possible partial loss of recirculation flow as the water temperature reaches saturation. While the plant may be tripped on a feedwater to steam flow ratio limit signal, it is assumed for this event that the plant is tripped on low drum level. The steam generator auxiliary feedwater flow will be initiated on a low drum level signal to prevent loop dryout.

B.1.2.13 U-13 Feedwater Throttle Valve Failed Open

This event is assumed to involve the feedwater throttle valve for one loop failing in the open position with the NSSS at the 40% power point. It is assumed that operator action does not correct the increasing drum level and that a high drum level signal causes automatic closure of the drum feedwater inlet isolation valve and a reactor trip.

B.1.2.14 U-14 Loss of One Recirculation Pump

Each steam generator is equipped with one recirculation pump which circulates water from the steam drum into the evaporators providing a recirculation ratio of 2:1 at full loop power. This event assumes instantaneous stoppage of the recirculation pump in one loop. The event includes loss of power to the pump, shaft seizure, and other pump malfunctions resulting in loss of forced recirculation through the evaporators. The plant trips based
47| on high evaporator outlet sodium temperature. The plant experiences transients similar to U-1b with the exception that the evaporators will experience an up transient as a result of the reduced water side flow.

B.1.2.15 U-15 Turbine Trips

B.1.2.15.1 U-15a Turbine Trip (Without Reactor Trip)

This event assumes that the turbine is tripped from full power (turbine throttle valve closes instantaneously) without a reactor trip. An 80% steam dump (bypass) to the condenser is provided which opens so that flow through the steam dump increases linearly from 0 to full flow in 3 seconds. Steam flow to all closed feedwater heaters is terminated with closure of the turbine throttle valves. Steam is admitted to the deaerating heater to maintain pressure in the deaerator above 100 psig. Reactor power, sodium flow, and steam bypass flow are ramped down at 3%/minute to 40% power. For design purposes, this ramp is followed by a normal shutdown to hot standby. A rapid turbine-valving transient is considered to be less severe than a turbine trip (without reactor trip) and, as a result, is not considered as a separate design transient.

B.1.2.15.2 U-15b Turbine Trip with Reactor Trip (Loss of Main Condenser or Similar Problems)

Turbine trip is assumed to occur when the main condenser, and thus the turbine dump (bypass) is unavailable. This causes the superheated steam flow to decrease to zero initially. The steam system pressures then increase and the superheated outlet power relief and safety valves open, returning the steam flow to about 100%. A reactor scram will be initiated based on turbine trip coincident with loss of condenser. The sodium pumps coast down, and steam flow and pressure are reduced. The steam generators see down transients in sodium temperature similar to U-1b, Reactor Trip From Full Power.

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B.1.2.18 U-18 Loss of All Offsite Power

Loss of main sodium pump motor power occurs and sodium flow will decrease to pony motor (driven by emergency standby power) flow. A reactor 47 trip is initiated and a turbine trip follows.

Since neither the feedwater pumps, recirculation pumps, nor the 47 cooling water pumps for the main condenser are on emergency (standby) power, the turbine bypass valve will remain closed (based on a signal noting loss of condenser cooling water) and steam system pressures will increase until the 47 superheater outlet and drum vent valves open, reestablishing steam flow.

The Auxiliary Feedwater Supply (AFS) portion of the Steam Generator 47 Auxiliary Heat Removal System (SGAHRS) will be initiated on low drum level. Pumps will take suction side flow from the protected storage tank to maintain drum level. Sufficient stored water is available in the protected tank to 58 make up for water lost to remove decay and stored heat by venting steam until the Protected Air Cooled Condenser System (PACCS) is capable of removing the 47 entire shutdown heat load.

B.1.2.19 U-19 Plant Shutdown in Response to Small Sodium - Steam/Water Leak Indication

This transient describes plant shutdown and affected loop depressurization for those sodium leak indications where immediate isolation and blowdown are not considered necessary but where a normal shutdown sequence is considered to be too slow.

The operational sequence will first reduce the system temperature differentials so that the thermal transients resulting from blowdown and dryout of the affected loop will be reduced. Initial power and flow reductions of a few percent per minute may be satisfactory. However, for defining this transient, an initial reactor trip is assumed, followed by affected module isolation and blowdown/dump. Steam system pressures and temperatures will then be reduced at normal shutdown rates until refueling conditions are reached.

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B.1.2.20 U-20 Turbine Bypass Valve

B.1.2.20.1 U-20a Inadvertent Opening of One Turbine Bypass Valve

From initial operation at 40% power, it is assumed that one turbine bypass valve (capacity of 20% of full flow) is fully opened. An overcooling of the steam generators will occur initially due to the increased steam flow. A trip on steam-feed mismatch occurs. Due to the failure of the bypass, there will be an excess of steam flow compared to intermediate sodium flow. This results in continued overcooling until the isolation valves are closed manually in 5 minutes.

B.1.2.20.2 U-20b Turbine Bypass Valve Fails Open Following Reactor Trip

This transient is included as it is representative of the transients that can rapidly blowdown and cool the steam generating system. Following reactor and turbine trip, the steam bypass system is used to maintain correct steam pressures and flows. Failure of a valve in this system in the open direction causes excessive steam flow with decreasing steam generator pressures and temperatures. The feedwater system will supply adequate water to maintain steam drum level. It is assumed that after 5 minutes operator action results in closure of the bypass valve.

B.1.2.21 U-21 Inadvertent Opening of Safety/Power Relief Valves

B.1.2.21.1 U-21a Inadvertent Opening of Evaporator Outlet Safety/Power Relief Valves

The valves are assumed to open instantaneously and this is accompanied by automatic closing of the inlet isolation valve of the affected evaporator. Pressure from the steam drum closes the check valve in the evaporator outlet line. This valve serves to isolate the evaporator at the outlet line and avoids blowing down the steam drum. The plant is tripped on high evaporator
47 outlet temperature.

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B.1.2.21.2 U-21b Inadvertent Opening of Superheater Outlet Safety/Power Relief Valves

For initial conditions of 390 MWt and a minimum decay heat, a full steam relief at a superheater exit is assumed. Loop steam flow will increase by 60% of full flow and turbine steam flow will decrease by 13%. The affected loop drum will initially decrease but feedwater flow will increase to maintain normal drum level. A reactor trip is assumed to occur from feed-steam flow mismatch and the transient is terminated by an operator shutting the superheater inlet valve 5 minutes after the trip.

B.1.2.22 U-22 Inadvertent Opening of SGAHRS Steam Drum Vent Valve

58 Inadvertent opening of a SGAHRS vent valve at the steam drum is assumed. The plant will trip on low steam/feedwater flow ratio, and the drum will depressurize. Depressurization may cause cavitation at the recirculation pump suction. The event is characterized by a decrease in recirculating water temperature and evaporator Na outlet temperature, and is terminated by operator action in isolating the SGAHRS steam drum vent line flow ten minutes after the plant trip.

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B.1.2.23 U-23 Inadvertent Opening of Evaporator Inlet Dump Valve

Both series valves are assumed to open instantaneously and this is accompanied by automatic closing of the inlet isolation valve of the affected evaporator. Pressure from the steam drum closes the check valve in the evaporator outlet line. This valve serves to isolate the evaporator at the outlet line and avoids blowing down the steam drum. The plant is tripped on high evaporator outlet temperature.

B.1.2.24 U-24 Reactor Trip with Failure of One PACC to Perform

The same operational sequence of event U-1a is assumed except the PACC does not start in one loop due to an assumed control logic failure.

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B.1.3 Emergency Events

All emergency events, which result in a reactor trip, shall be considered to result in a transient followed by a cooldown to refueling.

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B.1.3.1 E-1 Primary Pump Mechanical Failure

The event involves an instantaneous stoppage of the impeller of one primary pump while the system is operating at 100% power. The failure may be a seizure or breakage of the shaft or impeller. Primary system sodium flow in the affected loop decreases rapidly to zero as the pumps in the unaffected loops seat the check valve (thereby causing a check valve slam). A reactor trip will be initiated by the primary-intermediate flow ratio subsystem. Sodium flow in the intermediate circuit of the affected loop decays as in a reactor trip from full power (U-1b), modified by changes in natural circulation head. The event is characterized by a down transient in the hot leg of the intermediate circuit of the affected loop and by a check valve slam in the primary circuit of the affected loop.

B.1.3.2 E-2 Intermediate Pump Mechanical Failure

The impeller of one of the intermediate system pumps is assumed to stop instantaneously causing the flow in that circuit to decrease rapidly. A reactor trip is initiated by the primary to intermediate sodium flow ratio subsystem and the normal trip transient sequence is followed thereafter. The event is characterized by an up transient in the primary cold leg of the affected loop and by down transients in the steam generator modules of the affected loop since intermediate flow is limited to that produced by natural circulation.

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B.1.3.5 E-5 Loss of Primary Pump Pony Motor with Failure of the Check Valve to Shut

The initial conditions for this event are the same as for the pony motor failure upset condition (U-8) except that the check valve in the affected loop remains fully open. Flow will reverse in the affected loop while intermediate sodium flow remains constant. Thermal driving head in the affected loop is not considered. This condition will cause an abnormal temperature distribution in the IHX, a rapid decrease in the hot leg temperature in the affected loop and increasing core temperature due to the bypassed flow.

B.1.3.6 E-6 Design Basis Sodium/Water Reaction Event

The event begins with a small water/steam leak, from a steam generator tube, which escapes operator action, raises the intermediate heat transport system pressure close to the SWRPRS set point and causes localized overheating of another tube to the point of pressure rupture. The rupture area is equivalent to a double ended guillotine tube failure and produces a water/steam jet which causes localized overheating of a second tube to the point of pressure rupture after one second. The second rupture also equals one EDEG tube failure. The scenario repeats to the point of a third and final pressure rupture equalling one EDEG tube failure after another second.

The first equivalent double ended guillotine tube failure causes rupture disc actuation, followed by automatic isolation and blowdown of the water/steam side of both the evaporators and the superheater in the affected loop. In addition the reactor, the sodium pumps and the turbine are tripped. For the unaffected loops, the event is similar to a reactor trip from full power. Decay heat removal is maintained through the two remaining loops.

The event is classified as faulted for the affected Steam Generator unit, for the interconnecting piping between the affected steam generator unit and the associated rupture disc, for the super heater-to-evaporators sodium piping, and for the injected reaction products separation tank(s). For the rest of the loop, the occurrence is classified as an Emergency event.

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B.1.3.7 E-7 One Loop Natural Circulation Heat Rejection from Initial Two Loop Operation

Initial operation at a reduced power of 650 MWt is assumed, using Loops A and B. Loop C is assumed to be shutdown. Upon loss of forced sodium circulation, the reactor and turbine are tripped and a transient similar to E-16 occurs in Loops A and B. However, an additional assumption is made that the flow control valve between the turbine driven feed pump and the steam generator does not open on one of the operating loops. The resultant loss of makeup water causes the steam drum to boil dry; thus, the heat rejection capability of the loop is lost, and only one loop is available for the reactor decay heat removal.

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B.1.3.8 E-8 Rupture Disk Failure in SGS Sodium-Water Protection System

Flow of IHTS sodium through the failed rupture disks will initiate plant trip and activation of the steam generator water side blowdown system as in event U-11. Both evaporator modules and the superheater in the affected loop will be automatically isolated and blown down. It is assumed that the sodium rapid dump system is not manually activated.

B.1.3.9 E-9 Water/Steam Side Isolation and Dump of an Evaporator/Superheater Module with Failure of an Inlet or Outlet Isolation Valve to Close

B.1.3.9.1 E-9a Superheater Outlet Isolation Valve

If a superheater outlet isolation valve fails to close, the transient is essentially the same as U-11 since a check valve is installed in series with the isolation valve and the check valve stops backflow from the main steam header.

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B.1.3.9.2 E-9b Evaporator Inlet Isolation Valve

Failure of an evaporator inlet isolation valve to close will result in the affected loop being blown down to the water drain tank (and then to atmosphere). Turbine steam flow will be quickly reduced by approximately one-third.

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It is assumed that feedwater flow will not be capable of maintaining the drum level and that low drum level will be reached. The affected loop feedwater and auxiliary cooling system isolation valves will be closed on a low drum pressure signal and a reactor trip will occur. The water loss through the dump valve is assumed to be greater than SGAHRS flow and a dryout of the affected loop will occur.

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Amend. 58
Nov. 1980

B.1.3.9.3 E-9c Superheater Inlet Isolation Valve

Failure of a superheater inlet isolation valve to close will result in the affected loop steam flow being blown through the superheater exit power relief valve and safety valves to the atmosphere. The turbine steam flow will be reduced by approximately one-third. It is assumed that the loop feedwater can maintain the steam dump flows and the plant will continue operation without appreciable blowdown.

It is assumed the plant will be manually scrammed to conserve water or to permit blowdown if the operator is attempting to reduce the pressure in a leaking module.

B.1.3.10 E-10 Water Side Isolation of an Evaporator Module with Failure of the Water Dump Valve to Open

This transient assumes the same conditions as U-19a except the water dump valve at the evaporator module inlet fails to open. The module water/steam flow will stop and the input heat will raise the pressure until the module outlet
58 power relief valves open, drying the unit at a pressure (instead of a dump and dryout at low pressure).

For the unaffected loops, the event is similar to a reactor trip from full power. Decay heat removal is maintained through the unaffected steam
58 generator modules.

B.1.3.11 E-11 Steam Side Isolation of a Superheater with Failure of One Relief Valve to Open

This transient assumes the superheater isolation valves have closed as in U-11c in response to an attempt to blow the units down. However, due to a delay in opening time or failure of one of the power relief valves to
58 open, the pressure in the superheater reaches 1900 psig for 5 seconds before sufficient valves open to blow the units down. The transient is assumed to be the same as U-11c for the remainder of the system.

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B.1.3.14 E-14 Inadvertent Dump of Intermediate Loop Sodium

Inadvertent multiple dump valve openings are assumed. The affected loop IHTS sodium pump is assumed to be tripped on low pump tank level to avoid damage to the moving parts. This will cause a reactor trip.

47 The loop is assumed to be completely drained in 20 minutes. Loss of heat input will result in a down transient in the affected loop steam generator. For the unaffected loops, the event is similar to a reactor trip from full power. Decay heat removal is maintained through the two remaining loops.

B.1.3.15 E-15 DHRS Activation 24 Hours After Scram

58 This event postulates that none of the main heat transport loops are available for decay heat removal. It is also assumed that the plant has been tripped and 24 hours later, with the system temperatures brought to the hot standby temperature of 640°F, the DHRS is activated. The resulting system temperatures following initiation of decay heat removal system DHRS are
58 based on plant operation at 975 Mwt power prior to scram, and maximum decay power. Primary system sodium temperatures are based on the heat capacity of the reactor, overflow tank and two primary loops. At least one primary pump is assumed to be operating at pony motor speed to assume mixing of the primary sodium.

Intermediate system sodium temperatures are based on the heat capacity of the reactor, overflow tank, two primary loops and one intermediate loop. The event is terminated at refueling temperature.

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B.1.3.16 E-16 Three Loops Natural Circulation

From initial conditions of full power operation, complete loss
47 58 of forced sodium circulation in all loops is assumed. A reactor/turbine trip is initiated by primary pump under-voltage relays. Steam is relieved through the power operated relief and/or safety valves. Sodium pumps coast down and stop, and natural circulation flow is established in all sodium loops.
58 Auxiliary feedwater flow is established from the auxiliary feedwater portion of the steam generator auxiliary heat removal system based on low drum level signals. The turbine driven auxiliary feed pump takes suction from the protected water storage tank to maintain drum levels.

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Amend. 58
Nov. 1980

B.1.3.17 E-17 Two Loop Natural Circulation Heat Rejection from Initial Three Loop Operation

From initial conditions of full power (975 MWt) operation complete loss of forced sodium circulation in all loops is assumed. A reactor trip is initiated by primary pump under-voltage relays, and the E-16 transient occurs in all loops. In addition, the assumption is made that the flow control valve between the turbine driven feed pump and the steam generator does not open for one loop. The resultant loss of makeup water causes the steam drum to boil dry; thus, the heat rejection capability of one loop is lost, and only two loops are available for reactor decay heat removal.

B.1.3.18 E-18 Two Loop Natural Circulation

From initial two loop operating conditions at reduced power of 650 MWt, complete loss of forced sodium circulation is assumed. A reactor trip is initiated by primary pump under-voltage relays. Steam is relieved through the power operated relief and/or safety valves in the operating two loops. Sodium pumps in the two operating loops coast down and stop, and natural circulation flow is established in two sodium loops. Auxiliary feedwater flow to the two operating steam drums and SGS loops is established from the auxiliary feedwater portion of the steam generator auxiliary heat removal system based on low drum level signals. The turbine driven auxiliary feed pump takes suction from the protected water storage tank to maintain drum levels.

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B.1.3.19 E-19 Loss Of Flow in Two Sodium Loops

A failure is assumed causing all primary and intermediate sodium pumps to trip. The primary and intermediate sodium pumps for two loops coast down and stop. The primary and intermediate sodium pumps in the unaffected loop coast down to pony motor speed. Additional primary flow is provided by natural circulation. A pump electrics primary control rod trip and a steam to feedwater flow mismatch secondary control rod trip are initiated. During the transient prior to the scram, feedwater throttle control is assumed to act to maintain adequate feed flow for after scram conditions assuming loss of one feedwater pump.

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B.1.4 Faulted Events

B.1.4.1 DELETED

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B.1.4.2 F-2 DHRS Activation Without SGS Cooldown

This event occurs subsequent to the postulated loss of all heat transfer through the IHX's at the time of reactor trip from 3 loop rated power operation. At one half hour after scram, with the sodium in the reactor and HTS at approximately 1050°F, the DHRS is fully operational and assumes the decay heat load. Primary flow from 3 main coolant pumps at pony motor speed and maximum decay power is assumed. Primary system temperatures are based on the heat capacity of the reactor, overflow tank as appropriate for the overflow and return flow rates and all three primary loops. The criteria for accomodation of this event is that the coolant and DHRS boundaries remain intact and that the primary pumps retain the ability to function at pony motor speed. Active DHRS components must also retain their capability to perform their intended DHRS function. 22

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F-3 Feedwater Line Rupture

Components and systems shall be designed so that in the event of any of the following faulted events, sodium and steam generator system water/steam boundaries shall maintain their structural integrity (with the exception of the initiating failure).

F-3a Feedwater Line Rupture Between Steam Drum and Inlet Isolation Valve

This event assumes a rupture of a feedwater line between the inlet isolation valve and the steam drum. The result of the event is blowdown of the steam drum in the affected loop, interruption of steam flow through the superheater in the affected loop (due to check valve closure), and attendant up transients within the module. The steam generator in the affected loop dries out and is not available for decay heat removal. A stoppage of superheated steam flow in the affected loop occurs shortly after the rupture, due to reduced pressure in the steam drum. A plant trip occurs based on the steam-feedwater flow ratio. A low pressure signal from the steam drum protection subsystem results in closure of the feedwater isolation valves in the affected loop. This avoids excessive loss of steam plant water or protected storage water inventory. The up transient in the cold leg of the IHTS will propagate to an up transient in the cold leg of the PHTS.

F-3b Feedwater Line Rupture in Main Incoming Header

The reduction in feedwater line pressure will cause the feedwater line check valves to close simultaneously in all three drums. A reactor trip will be initiated on steam-feedwater flow ratio. The steam generator auxiliary feedwater flow will be initiated in all three loops on a low drum level signal to prevent loss of all plant cooling. The results of this event are similar to those for event U-5b, Loss of Feedwater Flow to All Steam Generators, and thus this event is evaluated as part of event U-5b.

F-4 Steam Line Rupture

These events postulate ruptures of the piping in the steam lines. The events are also postulated to insure that the steam generators and supports are capable of withstanding the reaction forces from the rupture without propagating failures to the units themselves.

Components and systems shall be designed so that in the event of any of the following faulted events, sodium and steam generator system water/steam boundaries shall maintain their structural integrity (with the
43 exception of the initiating failure).

F-4a Saturated Steam Line Rupture

A saturated steam line rupture is postulated between the steam drum outlet and the superheater inlet isolation valve. An immediate loss of superheater steam cooling occurs, and the superheater will later become isothermal at approximately the initial IHTS hot leg temperature. A reactor trip will occur due to steam-feedwater flow ratio. Depressurization of the loop will occur, and the superheater outlet steam check valve will close. The turbine steam flow will be reduced by approximately one-third. It is assumed that feedwater flow into the affected drum will not increase as greatly as steam flow from the affected drum, and the drum level will fall.

A low pressure signal from the steam drum protection subsystem closes the feedwater and auxiliary feedwater isolation valves. A dryout of the affected loop will occur.

F-4b Main Steam Line Rupture

A steam line rupture is postulated to occur between the manifold where the three loop superheated steam lines join together and the main steam line isolation valve. The turbine admission valve will rapidly close and the turbine will trip. Since the pressures have dropped, the turbine bypass will not open. Feedwater flow will increase rapidly through the units but will initially be unable to equal the blowdown flowrate. The superheater outlet isolation valve in each loop will be closed on a combined low superheater outlet pressure with SGAHRS initiation (due to high steam to feedwater flow ratio) signal. Reactor trip will occur on steam-feedwater flow ratio. Steam
58 system pressure will stabilize at the SGAHRS system vent valve settings. The steam generator auxiliary feedwater flow will be initiated on a low drum level signal to prevent loop dryout. Since this transient affects all three steam loops, its consideration is important to assure continuous SGS cooling can be maintained.

F-4c Rupture Between Superheater Module Outlet and Superheater Outlet Isolation Valve

A break in this location will cause an immediate increase in superheater steam cooling, loop depressurization, superheater outlet check valve closure, and reduction of turbine steam flow by approximately one-third. It is assumed that feedwater flow will not increase as greatly as steam flow from the affected drum and drum level will fall. A reactor trip will occur on steam-feedwater flow mismatch. Since the break is conservatively assumed to occur upstream of the superheater steam flowmeter, the affected loop superheater is not isolated on a high flow signal from the flowmeter. A low drum pressure signal closes the superheater inlet isolation valves, the steam drum drain valve and the feedwater isolation valve. The evaporator and drum
43 pressures will increase, opening relief valves. SGAHRS will be initiated on low steam drum level and cooling flow will be established for the affected
58 loop.

F-4d Rupture Between Superheater Outlet Isolation Valve and Main Steam Line

This event is similar to events F - 4a, b and c with the major difference being one of rate. For the unaffected loops, the event is essentially the same as event F - 4b, Main Steam Line Rupture, but steam flow rate is somewhat reduced. For the affected loop, the blowdown rate is slightly higher than that for F - 4b but not quite as rapid as that for F - 4a, Saturated Steam Line Break, and F - 4c, Rupture Between Superheater Module Outlet and Superheater Outlet Isolation Valve.

F-5 Recirculation Line Breaks

Components and systems shall be designed so that in the event of any of the following faulted events, sodium and steam generator system water/steam boundaries shall maintain their structural integrity (with the exception of the initiating failure).

F-5a Recirculation Line Break Between Drum and Recirculation Pump Inlet

The reactor is tripped on steam-feedwater flow ratio. The recirculation pump loses suction pressure resulting in loss of flow to the evaporators. Steam flow to the superheater also stops. The steam generator sodium temperatures initially decrease but then increase to the hot leg value. The steam generator blows down to zero gage pressure.
58

F-5b Recirculation Line Break Between Evaporator Outlet and Drum Inlet

A break at this location will initially overcool one or both
58 evaporator module due to the blowdown flow. The blowdown will cause the superheated steam flow check valve to close in the affected loop, initiating a reactor trip on steam-feedwater flow ratio. The affected loop will be isolated at the drum feedwater inlet and SGAHRS feedwater inlet and will
58 blow dry. After possible initial temperature decrease, the evaporator module temperatures will increase toward initial IHTS hot leg sodium temperature.
43

F-6 Intermediate Loop Sodium-Air Leak

A sodium leak to air is assumed to occur in one intermediate loop location outside of containment. A conservative assumption is made that no operator action is taken to trip the IHTS pump in the leaking loop or to drain
35 the loop. A reactor trip is caused by a PPS signal from the IHX outlet temperature sensors. For the unaffected loops, the event is similar to a reactor trip from full power. Decay heat removal is maintained through the two
58 remaining loops.

B.2 SELECTION OF UMBRELLA TRANSIENTS

Structural evaluation of the effects of each individual duty cycle event on each reactor plant component would present an unnecessary and a prohibitively expensive and time consuming effort. To avoid such a massive and unnecessary effort, duty cycle events have been grouped for each component to permit evaluating the effects of a single transient event (umbrella) which is conservatively representative of the group. The duty cycle events were grouped separately for each component since the transient effects of any particular duty cycle event are not necessarily the same for each component.

The procedure employed to select a set of umbrella transients for each component took the following steps:

A. The first step was to characterize the expected transient for each duty cycle event to the greatest extent possible. For the normal events, this resulted in most events being directly applicable with only a small amount of grouping possible due to the general dissimilarity of these events. For the upset and emergency duty cycle events, the characterization included a tabulation specifying the following variables for each duty cycle event:

   a) Initial power

   b) Decay heat level (normal or minimum)

   c) Transient terminal condition (hot standby or refueling)

   d) Initial conditions of temperature, pressure, and flow

   e) Characterization of resulting transient as to general temperature pattern and approximate magnitude.

B. The next step was to arrange the upset and emergency events in groups which were similar in transient effect. Each duty cycle event was placed in a matrix location depending on:

   a) The initial flow condition (40 or 100%)

   b) The final state of the plant (hot standby or refueling)

   c) When applicable, which sodium system received the major transient effect of the event (primary or intermediate)

   d) When applicable, whether the IHX inlet or outlet of the affected component received the major transient effect of the event

e) Whether the initial effect of the event resulted in an up or down temperature ramp.

C. The events in each group were then reviewed to determine which event provided the most severe transient and could be used as an umbrella for the other events. In some cases, more than one event in a particular group was selected as a potential umbrella event. This was generally done when it was judged that there might be significant differences between two events and, therefore, it was desired to know the specific effects of each event.

D. The specified frequency of each transient umbrella group is determined by summing the frequencies of the individual duty cycle events assigned to that group. Since the duty cycle event used for the umbrella event for that group provides the most severe transient affect for that group, this approach results in a conservative design when applied by a component vendor to evaluate structural damage.

TABLE B-1

PRELIMINARY DESIGN DUTY CYCLE EVENT FREQUENCIES

Event		Frequency
1. Normal Events		
N-1	Dry system heatup and cooldown, sodium fill and drain	5 total system + 8 per loop + 17 additional for entire intermediate loop exclusive of IHX
N-2a	Startup from refueling	140
N-2b	Startup from hot standby	700
N-3a	Shutdown to refueling	60
N-3b	Shutdown to hot standby	210
N-4a	Loading and unloading	9300 (loading) 9300 (unloading)
N-4b	Load fluctuations	46500 (up) 46500 (down)
N-5	Step load changes of ± 10% of full load	750 (+10%) 750 (-10%)
N-6	Steady state temperature fluctuations	30×10^6
N-7	Steady state flow induced vibrations	10^{10} (sodium)
2. Upset Events		
U-1a	Reactor trip from full power with normal decay heat	180(1)
U-1b	Reactor trip from full power with minimum decay heat	180(1)
U-1c	Reactor trip from partial power with minimum decay heat	180(1)
U-2a	Uncontrolled rod insertion	10
U-2b	Uncontrolled rod withdrawal from 100% power	10

(1) - The total frequency for U-1 is associated with normal decay heat from full power so as to balance the trips associated with partial decay heat for events U-2 through U-23.

TABLE B-1 (Continued)

	Event		Frequency
	U-2c	Uncontrolled rod withdrawal from startup with automatic trip	17
47	U-2d	Uncontrolled rod withdrawal from startup to trip point with delayed manual trip	3
	U-2e	Plant loading at max. rod withdrawal rate	10
48 47	U-2f	Reactor startup with excessive step power change	50(2)
	U-3a	Partial loss of primary pump	2 per loop
	U-3b	Loss of power to one primary pump	5 per loop
	U-4a	Partial loss of one intermediate pump	2 per loop
	U-4b	Loss of power to one intermediate pump	5 per loop
	U-5a	Loss of AC power to one feedwater pump motor	10
	U-5b	Loss of feedwater flow to all steam generators	5
45			
	U-7a	Primary pump speed increase	5
	U-7b	Intermediate pump speed increase	5
	U-8	Primary pump pony motor failure	5 per pump
	U-9	Intermediate pump pony motor failure	5 per pump
	U-10a	Evaporator module inlet isolation valve closure	4 per loop
	U-10b	Superheater module inlet isolation valve closure	2 per loop
	U-10d	Superheater module outlet isolation valve closure	2 per loop

47 (2) - These events are part of the startups specified for event N-2b and should not be added as separate startups.

TABLE B-1 (Continued)

Event			Frequency
	U-11a	Water side isolation and dump of both evaporators and the superheater	6 per loop
	U-11b	Water side isolation and dump of evaporator module	6 per loop
	U-11c	Water side isolation and dump of superheater	3 per loop
47	U-12	Loss of feedwater flow to one steam generator loop	3 per loop
	U-13	Feedwater throttle valve failed open	6 per loop
	U-14	Loss of one recirculation pump	8 per loop
	U-15a	Turbine trip (without reactor trip)	50
	U-15b	Turbine trip with reactor trip (loss of main condenser or similar problem)	10
58			
35			
	U-18	Loss of all offsite power	16
35	U-19	Plant shutdown in response to small sodium-steam/water leak indication	9 per loop
47			
	U-20a	Inadvertent opening of one turbine bypass valve	5
	U-20b	Turbine bypass valve fails open following reactor trip	5
	U-21a	Inadvertent opening of evaporator outlet safety/power relief valves	5 per loop

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TABLE B-1 (Continued)

	Event		Frequency
	U-21b	Inadvertent opening superheater outlet safety/power relief valves	3 per loop
58 47	U-22	Inadvertent opening SGAHRS Steam Drum Vent Valve	3 per loop
	U-23	Inadvertent opening of evaporator inlet dump valve	3 per loop
47	U-24	Reactor trip with failure of one PACC to perform	10(1)
3.	Emergency		
	E-1	Primary pump mechanical failure	Each component must accommodate 5 occurrences of the most severe emergency transient for that component (one every 6 years) plus two consecutive occurrences of the most severe event (or consecutive occurrences of two unlike events if the unlike events provide a more severe effect than consecutive occurrences of the most severe event).
	E-2	Intermediate pump mechanical failure	
43			
47	E-5	Loss of primary pump pony motor with failure of the check valve to shut	
47	E-6	Design basis steam generator sodium-water reaction	
	E-7	One loop natural circulation heat rejection from initial two loop operation	
47	E-8	Rupture disc failure in SGS sodium-water protection system	

(1) - These events are part of the reactor trips for event U-1a and should not be added as separate trips.

47

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TABLE B-1 (Continued)

Event		Frequency
E-9a	Water/steam side isolation and dump of an evaporator/superheater module with failure of module outlet isolation valve to close	
E-9b	Water/steam side isolation and dump of an evaporator superheater module with failure of an evaporator inlet isolation valve to close	
E-9c	Water/steam side isolation and dump of an evaporator/superheater module with failure of a superheater inlet isolation valve to close	
E-10	Water side isolation of an evaporator module with failure of the water dump valve to open	
E-11	Steam side isolation of a superheater with failure of one relief valve to open	
E-14	Inadvertent dump of intermediate loop sodium	
E-15	DHRS Activation 24 Hours After Scram	If event E-15 is the most severe condition for a component, it shall be evaluated for a frequency of 2 for that component in addition to the 7 occurrences of the next most severe transient.
E-16	Three Loop Natural Circulation	
E-17	Two loop natural circulation heat rejection from initial three loop operation	
E-18	Two loop natural circulation	
E-19	Loss of Flow in Two Sodium Loops	

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TABLE B-1 (Continued)

Event		Frequency
4. Faulted Events		
F-1	Deleted	
F-2	DHRS Activation Without SGS Cooldown	
F-3	Feedwater Line Ruptures	
F-3a	Feedwater Line Rupture Between Steam Drum and Inlet Isolation Valve	
F-3b	Feedwater Line Rupture in Main Incoming Header	
F-4	Steam Line Ruptures	
F-4a	Saturated Steam Line Rupture	
F-4b	Main Steam Line Rupture	
F-4c	Rupture Between Superheater Module Outlet and Superheater Outlet Isolation Valve	
F-4d	Rupture Between Superheater Outlet Isolation Valve and Main Steam Line	
F-5	Recirculation Line Breaks	
F-5a	Recirculation Line Break Between Drum and Recirculation Pump Inlet	
F-5b	Recirculation Line Break Between Evaporator Outlet and Drum Inlet	
F-6	Intermediate Loop Sodium-Air Leak	

TABLE B-2

CRBRP DUTY CYCLE

PRELIMINARY UMBRELLA TRANSIENT SUMMARY FOR UPSET AND EMERGENCY EVENTS

Event Number	Event Description	Initial Power %	Decay Heat Condition	Terminal Conditions	Reactor Vessel	IHX	Primary Pump	Check Valve	Intermediate Pump	Evaporator	Superheater
U-1a	Reactor Trip from Full Power with Normal Decay Heat	100	Normal	Hot Standby	X	X	X				
U-1b	Reactor Trip from Full Power with Minimum Decay Heat	100	Minimum	Hot Standby	X	X	X	X	X	X	X
U-2a	Uncontrolled Rod Insertion	100	Minimum	Hot Standby		X	X			X	X
U-2b	Uncontrolled Rod Withdrawal from 100% Power	100	Minimum	Hot Standby	X	X	X	X		X	X
U-2d	Uncontrolled Rod Withdrawal from Startup to Trip Point (With Delayed Manual Trip)	0	Minimum	Hot Standby	X	X	X			X	X
U-2e	Plant Loading at Maximum Rod Withdrawal Rate	40	Not Applicable	100	X	X	X	X	X	X	X
U-4a	Intermediate Pump Control Failure	100	Minimum	Hot Standby		X					
U-11a	Water Side Isolation and Dump of Both Evaporators and the Superheater	100	Minimum	Refueling	X	X		X	X	X	X
U-11b	Water Side Isolation and Dump of an Evaporator Module	100	Minimum	Refueling						X	X
U-16	Operating Basis Earthquake	100	Minimum	Hot Standby	X	X	X	X	X	X	X
U-18	Loss of all offsite power	100	Normal	Hot Standby	X	X	X	X	X	X	X
U-20b	Turbine Bypass Valve Fails Open Following Reactor Trip	100	Minimum	Hot Standby		X	X		X		

35

47

TABLE B-2 (Continued)

Event Number	Event Description	Initial Power %	Decay Heat Condition	Terminal Conditions	Reactor Vessel	IHX	Primary Pump	Check Valve	Intermediate Pump	Evaporator	Superheater
U-21a	Inadvertent Opening of Evaporate Outlet Relief Valve	100	Minimum	Hot Standby						X	X
U-21b	Inadvertent Opening of Superheater Outlet Safety/ Power Relief Valves	40	Minimum	Hot Standby	X	X	X	X	X	X	X
U-23	Inadvertent Opening of Evaporator Inlet Dump Valve	100	Minimum	Hot Standby		X			X		
E-1	Primary Pump Mechanical Failure	100	Minimum	Refueling	X	X(Also as Upset)		X			
43											
E-5	Loss of a Primary Pump Pony Motor with Failure of the Check Valve to Shut	100	Minimum	Refueling	X	X	X			X	X
E-6	Design Basis Steam Generator Sodium-Water Reaction	100	Minimum	Refueling		X			X	X	X
E-7	One Loop Natural Circulation from Initial Two Loop Operation	100 per loop	1.25 x Normal	Refueling	X	X				X	X
43											
E-14	Inadvertent Dump of Intermediate Loop Sodium	100	Minimum	Refueling						X	X
35 E-16	Three Loop Natural Circulation	100	1.25x Normal	Hot Standby	X	X	X			X	X