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LIGHT-WATER-REACTOR COUPLED NEUTRONIC AND THERMAL-HYDRAULIC CODES

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ABSTRACT

An overview is presented of computer codes that model light water reactor cores with coupled neutronics and thermal-hydraulics. This includes codes for transient analysis and codes for steady state analysis which include fuel depletion and fission product buildup. Applications in nuclear design, reactor operations and safety analysis are given and the major codes in use in the USA are identified. The neutronic and thermal-hydraulic methodologies and other code features are outlined for three steady state codes (PDQ7, NODE-P/B and SIMULATE) and four dynamic codes (BNL-TWIGL, MEKIN, RAMONA-3B, RETRAN-02). Speculation as to future trends with such codes is also presented.

INTRODUCTION

The neutron flux, and hence the power, in a light water reactor (LWR) is (among other things) a function of the fuel pellet temperature and the water temperature and density. These thermal-hydraulic variables are in turn a function of the power. In many applications it is necessary to have this coupling explicitly represented throughout the core. The codes that have been developed to satisfy this requirement are of two general types.

The first type includes the effect of fuel depletion and fission product buildup and hence is useful for steady state calculations at different points in the life of a fuel cycle as well as for calculations of slow transients due to changing xenon concentration. These calculations have significance in fuel management, operations and safety. Codes for these applications which represent the core in three dimensions are frequently called core simulators. One dimensional codes will not be considered in this paper because of their limited use in the aforementioned applications.

The second type of code couples time dependent neutronics and thermal-hydraulics in order to determine core behavior under accident or operational transient conditions. This is a much more difficult analysis problem with fewer applications and hence there is less experience available. The codes that are used may be either one, two or three dimensional and are called core dynamics codes.

We consider each type of code separately in the following. For both core simulators and core dynamics codes we first discuss applications, and then

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identify the major codes in use in the USA. Characteristics of selected codes are then summarized. This includes the neutronics and thermal-hydraulics modeling as well as other features. Finally something is said about future trends relative to code development.

APPLICATIONS OF CORE SIMULATORS

Core simulators are used for incore fuel management (or equivalently for nuclear design). Fuel enrichment requirements, loading patterns, and cycle length can be determined with the help of these codes. For predictive calculations, assumed operating conditions are input to the code whereas for core-follow calculations the actual operating history is used. The codes are also used to help determine spent fuel composition.

Applications with operational significance are closely related to the above fuel management examples. Core simulators are useful for determining operating strategies for fuel preconditioning and for load following and in general for determining critical statepoints. These functions can be done at the plant site with fast running codes if the proper hardware is available. On-line monitoring with such a code also allows for the detection of anomalous behavior in the reactor.

Calculations with operational safety significance include those to determine the reactor protection system setpoints, the thermal margins as a function of different operating conditions (either directly or by supplying data to another code) and the differential and integral control rod (or bank) reactivity worths. Core simulators are also used to generate core-average delayed-neutron data and reactivity functions to describe control rod motion and thermal-hydraulic feedback, all of which are needed in safety calculations done with a point kinetics model.

MAJOR CORE SIMULATORS

Table 1 lists the major pressurized water reactor (PWR) and boiling water reactor (BWR) core simulators in use in the USA. The principal user and appropriate references are also given. Codes A.1-A.4 and B.1-B.2 are used by the five U.S. fuel vendors and do not have a wide circulation. PRESTO (Code A.5 and B.3) is a foreign product which has recently been made available to utilities in this country via the University Computing Co. The CORE code is a general LWR code, however, its application to BWRs is most notable.

The codes which have had the most widespread use by the utility industry are NODE-P/B (Codes A.6 and B.4), SIMULATE (Codes A.7 and B.5) and PDQ7 (Codes A.8) and they will be discussed further in the next section. NODE-P/B has many variants e.g. SUPERNODE-P/B is the equivalent Control Data Corp. product and NAI3D-P/B is the equivalent Nuclear Associates International product. SIMULATE is of particular interest because its support by the Electric Power Research Institute gives it the potential for an even larger number of users than it already has.

Although PDQ7 (or PDQ8) can be used as a core simulator for both PWRs and BWRs it has had no such application to BWRs and its use as a PWR simulator has been

TABLE 1
LWR Core Simulators

Code	Principal User	Reference
A. PWR Codes		
1 PALADON	Westinghouse (W)	C1
2 ROCS	Combustion Engineering (C-E)	O1,O2
3 FLAME 3	Babcock & Wilcox (B&W)	M1
4 XTG	Exxon Nuclear (ENC)	S1
5 PRESTO	Scandpower (ScP)	S2
6 NODE-P	Utilities, Brookhaven Nat'l Lab (BNL)	R1
7 SIMULATE	Yankee Atomic (YAEC), Utilities	V1
8 PDQ7, PDQ8	Bettis Lab, B&W, BNL	P1,P2
B. BWR Codes		
1 GEBS (PANACEA)	General Electric (GE)	W1
2 XTGBWR	Exxon Nuclear	E1
3 PRESTO	Scandpower	S2
4 NODE-B	Utilities, BNL	K1
5 SIMULATE	YAEC, Utilities, BNL	V1,V2
6 CORE	TVA	F2

limited. Its widespread use by the industry has been primarily for two dimensional (X-Y plane) calculations needed for fuel assembly or core calculations. In these applications there is generally no direct coupling between the neutronics and thermal hydraulics. Hence, PDQ7 is not an important core simulator (primarily because its finite difference algorithm makes computation time large for a detailed three dimensional calculation); nevertheless, it is an important reactor analysis code and hence is discussed below in further detail along with NODE-P/B and SIMULATE.

SUMMARY OF CORE SIMULATOR CHARACTERISTICS

Table 2 summarizes the characteristics of PDQ7, NODE-P/B and SIMULATE. In particular the neutronics and thermal-hydraulics methodology are outlined and information is given on the burnup calculation and other features of the code which are of interest to a core analyst. An important difference between PDQ7 and the latter two codes is the use of nodal methods by NODE-P/B and SIMULATE which allow them to be relatively fast running. SIMULATE in turn is slower running than NODE-P/B; a penalty imposed by the slightly more sophisticated modeling and the improved features summarized in Table 2. A basic difference between the two nodal codes is the use of one neutron energy group in NODE-P/B and two groups in SIMULATE. Since SIMULATE only treats the thermal group approximately, however, this is generally referred to as a 1-1/2-group method.

Both NODE-P/B and SIMULATE rely on user supplied correlations to obtain thermal-hydraulic variables of interest. Auxiliary calculations are therefore necessary. The use of input correlations extends to steam table information required by the calculation.

Table 2
LWR core depletion code characteristics

	PDQ7	NODE-P/B	SIMULATE
Neutron flux calculation			
Geometry	Flexible[a]	X,Y,Z: 1 node/assembly	X,Y,Z; 1 or 4 nodes/assembly
Symmetry	Mirror, rotational	Mirror, rotational	Mirror, rotational
Boundary condition	Fixed flux	Core-reflector leakage	Core-reflector albedo
Energy groups	1 - 5	One	1, 1-1/2
Solution algorithm	Finite differencing	FLARE-nodal method	Several nodal options
Searches	Control rod position, poison concentration	Soluble boron, Haling burnup	Control rod position, power, flow, soluble boron, Haling burnup
Depletion/Fisson products	Any user specified linear chain, e.g. Xe, Sm, actinides	Explicit I,Xe exposure void and control dependence history	Explicit I,Xe,Pm,Sm exposure and void history dependence
Thermal-hydraulics methodology			
Fuel temperature	Conduction model or correlation[b]	Proportional to power[b]	Quadratic correlation with power[b]
Flow distribution	Input; cross flow allowed in (R,Z) geometry	Quadratic correlation with power[b]	Linear correlation with power[b]
Coolant temperature/density	Uses vapor generation and slip correlation	PWR: temperature from quadratic correlation with quality.[b] Density from quadratic correlation with temperature.[b] BWR: void fraction from quality using slip correlated with exit quality and inlet velocity. Subcooled voids from correlation.	PWR: density from quadratic correlation with quality[b] BWR: same as NODE-B
Other features	Flux synthesis Adjoint flux Incore detector responses	Fuel shuffling Incore detector response[b] Normalization recommended CHFR calculated	Fuel shuffling Incore detector response[b] Normalization recommended BWR inlet subcooling, bypass voiding calculated
Availability[c]	NESC,UCC,CYERBNET	EPSC,UCC,CYBERNET	EPSC
References	P1,P2,S3,B1,R2	R1,R3,K1,B2,D1	V1,V2,V3

a Rectangular, cylindrical, spherical or hexagonal geometry in 1,2 or 3 dimensions with variable mesh size
b User must supply correlations
c NESC: National Energy Software Center; UCC: University Computing Company; CYBERNET: Control Data Corp; EPSC: Electric Power Software Center

Under "Other Features" in Table 2 it is noted that normalization of the calculation to measurements made at operating plants is recommended. Although this diminishes the mechanistic nature of the calculation it has proven to be important in enabling these codes to be successful as core simulators. This might not be the case if the codes were used for abnormal situations far removed from the operating conditions at which they were normalized.

FUTURE TRENDS WITH CORE SIMULATORS

Since current core simulators have a long history of successful application it seems unlikely that new codes will be written or that major neutronic or thermal-hydraulic modeling changes will be incorporated into existing codes. However, note the exceptions to this. An improved two-phase flow model (L1) is already being incorporated into SIMULATE. Improved methods (based on two-group nodal techniques) also exist for the neutronics and, for example, the method used in the QUANDRY code (G4) could be the basis for a new core simulator.

Instead of concentrating on totally new modeling there may be refinements to existing models and the incorporation of additional models from auxiliary codes into the core simulators. Examples of refinements are: making albedos a function of local water density and having the correction for thermal neutron diffusion dependent on the presence of control blades and void fraction. Examples of models that exist in auxiliary codes that might be incorporated include a thermal margin calculation for SIMULATE and a flow distribution calculation for NODE-P/B or SIMULATE.

Most of the future effect is expected to be in areas that directly affect the core analyst. Improving the linkage to other codes is very important (L2). This includes both the streamlining of the data generation process required for the core simulator and the automatic generation of data by the simulator for use in neutron kinetics calculations. An allied concern is improved documentation that provides the user with procedures for generating the data he needs. There is also an interest in computational speed - of particular interest in codes that may be used for on-line monitoring.

APPLICATIONS OF CORE DYNAMICS CODES

If there is a situation in which the power distribution changes rapidly in time i.e. the spatial and temporal changes are non-separable, then a code which uses spatial neutron kinetics must be employed - preferably with coupling to the thermal-hydraulics. For a PWR the most obvious situation in which this is necessary is a rod ejection accident (REA). For this accident a full three dimensional representation is necessary.

In most PWR transients initiated by perturbations in the thermal-hydraulics (as opposed to a reactivity initiated accident such as the REA) the changes in the core power distribution are relatively slow and spatial kinetics is not necessary. Exceptions to this rule can be found. If there is a failure of control rods to insert after a reactor trip initiated by some transient (an ATWS event or a stuck rod situation) then the core behavior becomes important and spatial kinetics may be necessary. Reference B10 demonstrates this for a transient initiated by a loss-of-feedwater. In a steam line break in a plant using a once-through steam generator the rapid depressurization of the primary and the lack of shutdown in an assembly due to a stuck-rod could cause thermal limits to be exceeded. However, the evidence as to whether spatial kinetics is needed (B10,N1) is inconclusive.

In the case of BWRs it is clear that because of the strong coupling between void and power there are many transients initiated by thermal-hydraulic perturbations in which spatial neutron kinetics coupled with a thermal-hydraulics calculation is necessary. This is certainly the case for ATWS events and for overpressurization transients. In the latter case reactor trip may be initiated only when the core is already experiencing a power surge. In these types of transients it is usually changes in the axial power shape that are significant - frequently the radial shape does not change much. In such a case a one dimensional neutron kinetics solution may be adequate. If there is an interest in following the hot channel or if there is only a partial control rod insertion then a three dimensional representation is necessary.

The reactivity initiated accident of concern in a BWR is a rod drop accident. As in the case of the REA in a PWR this requires a three dimensional core representation or at least an (R,Z) geometry.

MAJOR CORE DYNAMICS CODES

A list of coupled time dependent neutronic and thermal-hydraulic codes is given in Table 3 for application to both PWRs and BWRs. The principal user, appropriate references and the number of dimensions in the spatial neutron kinetics formulation is also given. Codes A.1-A.4, A.8, B.1 and B.2 are used by the five U.S. fuel vendors and, with the exception of MEKIN (A.3), are not expected to have further circulation. TITAN (A.5 and B.5) is a combination of the QUANDRY neutron kinetics code and the THERMIT thermal-hydraulics code and is still under development. TWIGL (A.6) has had limited PWR application and there is no documentation of any FX2-TH (A.7) applications. FIESTA (A.8) is for the special application of calculating scram reactivity. RETRAN-02 (A.9) also has not yet been applied to PWRs (with its spatial kinetics option) in a documented study.

TABLE 3
LWR Core Dynamics Codes

Code	Dimensions	Principal User[a]	Reference
A. PWR codes			
1 HERMITE	3	C-E	R4,R5
2 TWINKLE	3	W	B3
3 MEKIN	3	B&W, BNL	A1,B4,B5
4 XTRAN	2	ENC	E1
5 TITAN	3	MIT	G1,G2
6 TWIGL	2	Bettis,C-E	Y1
7 FX2-TH	2	ANL	S5
8 FIESTA	1	C-E	D2
9 RETRAN-02[b]	1	EPRI/EI	M2

TABLE 3 (cont.)
LWR Core Dynamics Codes

Code	Dimensions	Principal User[a]	Reference
B. BWR Codes			
1 ODYN[b]	1	GE	A2,S4
2 COTRAN	2	ENC	P1
3 RAMONA-3B[b]	3	BNL	D4
4 MEKIN	3	BNL	A1,B4
5 TITAN	3	MIT	G1,G2
6 BNL-TWIGL	2	BNL	C2,D3
7 RETRAN-02[b]	1	EPRI/EI,TVA	M2

a See Table 1 for definition of abbreviations. Also: ANL, Argonne; EI, Energy Incorporated
b Also models nuclear steam supply system.

Since the applications to BWR safety analysis are more extensive than to PWR analysis we consider the BWR codes in more detail. We eliminate consideration of the (proprietary) fuel vendor codes and the TITAN code with which there is limited experience. In the next section, therefore we consider BNL-TWIGL, MEKIN, RAMONA-3B and RETRAN-02.

SUMMARY OF CORE DYNAMICS CODE CHARACTERISTICS

Table 4 summarizes the important characteristics of the four codes of interest for BWR core dynamics. The following text comments on the entries (sequentially) in the table.

Each code has a different set of field equations for the two-phase water system. Although a mixture momentum equation is written for RAMONA-3B in the vessel it is integrated around the flow path resulting in one integrated momentum equation for each channel through the core. The RAMONA-3B and BNL-TWIGL formualtions are therefore similar by virtue of having a single pressure for the entire core.

BNL-TWIGL and RAMONA-3B are also similar in that they both use slip correlations to obtain relative phasic velocities. Specifically, RAMONA-3B has two options (Bankoff-Malnes and Bankoff-Jones) for this correlation and also uses a fixed bubble rise velocity whereas BNL-TWIGL uses the Bankoff-Jones correlation and no bubble rise velocity. The so-called dynamic slip model in RETRAN-02 is based on the drift flux formulation and was not available in the original version of the code. (Homogeneous flow or an algebraic slip are also available in RETRAN-02). Each code has a different subcooled boiling model indicated by the developer's name in the table.

RETRAN-02 with its 16 heat transfer regimes is clearly more sophisticated than the other three codes in this regard. Note also that RETRAN-02 is the only one of the four codes that allows for superheated steam and a metal-water reaction.

BNL-TWIGL has no need for a two-phase friction multiplier because it does not solve a momentum equation. Another consequence of this is that the flow distribution must be specified for BNL-TWIGL.

The calculation of critical heat flux is done in RAMONA-3B with the RELAP4/MOD7 correlation while with RETRAN-02 the user has several options. Both of these codes have specific models for post-CHF flow and heat transfer. Although BNL-TWIGL does allow for heat transfer with very high void fractions that model has not been fully validated for the post-CHF regime.

One of the most important distinctions between these codes is the fact that RAMONA-3B and RETRAN-02 model the entire nuclear steam supply system (NSSS) whereas MEKIN and BNL-TWIGL model only the core. The inlet flow rate, temperature and pressure therefore must be input into the latter two codes whereas for the codes that model the NSSS this is calculated by the code using boundary conditions elsewhere in the system.

Although RETRAN-02 is capable of calculating a flow distribution this is a moot point since it is a one-dimensional neutron kinetics model and hence it is most likely that only one thermal-hydraulic channel would be used to represent the in-channel flow.

The bypass channel can be represented in RAMONA-3B and RETRAN-02 in order to properly calculate the hydraulic conditions in the core. However, the water density in the bypass region is not used as feedback to the neutron cross sections for use in obtaining the neutron kinetics solution.

The conduction model limitation referred to for BNL-TWIGL is not serious and results in not knowing the temperature distribution through the pellet. For RAMONA-3B the limitation is that fuel rod properties (geometry, conductivity etc.) cannot be different in different channels.

A hot bundle representation is essential in a one dimensional core code such as RETRAN-02 and theoretically should not be necessary with a more detailed geometry such as is found in the other three codes. Nevertheless, there are situations when those codes might be used with only one dimension represented or with such a coarse radial mesh that a hot bundle calculation would be very useful.

A major distinction between the four codes with regard to neutron kinetics is the use of a finite difference algorithm in BNL-TWIGL and MEKIN as opposed to the use of a nodal method in RAMONA-3B and a space-time factorization method in RETRAN-02. The latter two approaches are meant to improve the computational speed. The coarse-mesh method in RAMONA-3B (related to the steady state algorithm used in PRESTO and SIMULATE) has proven to be relatively fast running. The fact that albedo boundary conditions are used rather than an explicit reflector also helps improve the computational speed of RAMONA-3B. The space-time factorization method used in RETRAN-02 has had successful application in FX2-TH and FIESTA but little experience has been collected to determine whether it will be more efficient for BWR applications than, for example,

TABLE 4
Core Dynamics Code Characteristics[a]

	BNL-TWIGL	MEKIN	RAMONA-3B	RETRAN-02
Core Thermal Hydraulics				
Field Eqns - number	3	3	4	3
Mass	Mixture	Mixture	Vapor, Mixture	Mixture
Energy	Liquid,Vapor	Mixture	Mixture	Mixture
Momentum	No	Mixture	Mixture	Mixture
Relative Phasic Velocities	Bankoff-Jones (BJ) Slip	Homogeneous	Bankoff-Malnes/ (BJ) Slip	Dynamic Slip
Subcooled Boiling	Lellouche	Levy	Bakstad	Lellouche/ Zolotar
Heat Transfer Regimes	3	2	4	16
Two-Phase Friction Multiplier	No	3 Options	Yes	3 Options
CHF Calculation	No	No	RELAP4/MOD7	4 Options
Post-CHF Modeling	Some	No	Yes	Yes
Superheated Steam	No	No	No	Yes
Inlet Conditions	Input	Input	Calculated	Calculated
Flow Distribution	Input	Calculated	Calculated	Calculated
Bypass Channel	No	No	Yes-Not coupled to neutronics	Yes-Not coupled to neutronics
Conduction Model Limitations	Only average pellet temp.	--	Only one rod type	--
Direct Energy Deposition	Clad & Coolant	Coolant	Coolant & Bypass	Coolant & Bypass
Metal-Water Reaction	No	No	No	Yes
Hot Bundle	No	No	No	Yes

a See text for explanation

TABLE 4 (cont.)
Core Dynamics Code Characteristics[a]

	BNL-TWIGL	MEKIN	RAMONA-3B	RETRAN-02
Neutron Kinetics				
Dimensions (Geometry)	2:(R,Z)[b]	3:(X,Y,Z)[b]	3:(X,Y,Z)[b]	1:(Z)
Energy Groups[c]	2	2	1-1/2	2
Delayed Neutron Precursor Groups[c]	6	6	6	6
Solution Algorithm	Finite Difference	Finite Difference	Nodal Method	Space-Time Factorization
Boundary Conditions	Zero Flux or current	Zero Flux, Albedo	Albedo	Zero Flux
Miscellaneous				
Decay Heat[d]	No	$F(t_0,t)$	$F[t_0,P(t),t]$	$F[P(t),t]$
Feedback Dependence[e]	$U,T_C,T_F,$	$U,T_C,T_F,$ N_X	$U,T_C,T_F,N_X,C_B,$ (E,U_E)	U,T_C,T_F
Reactivity Edits	Yes	Yes	No	No
Single or Multiple Rod Groups Per Channel	Multiple	Single	Single	Multiple
References				
Modeling	C2,D3,Y1	A1,B4,R6	B8,B9,D4	M2,M3
Application/ Qualification	C3,C4	B7,C5	B6,D4	F1,G4

a See text for explanation
b Reducible to one dimension
c Maximum number
d Pre-transient operating time, t_0; time, t; instantaneous power, P(t).
e Moderator density, U; coolant temperature, T_C; fuel temperature, T_F; xenon number. density, N_X; boron concentration, C_B; exposure, E; void history, U_E.

a finite difference method. Note also that RAMONA-3B uses a 1-1/2 group method while all the other codes use a two group method.

Three of the codes have models for decay heat whereas BNL-TWIGL which was originally designed for fast transients has none. The models in RAMONA-3B and RETRAN-02 are suitable for the analysis of ATWS events because the decay heat is a function of instantaneous power whereas the decay heat model in MEKIN is for shutdown situations.

All four codes allow for cross sections to be a function of the three thermal-hydraulic variables moderator density and temperature and fuel temperature; although in most BWR transients the dependence on moderator temperature can be neglected. RAMONA-3B and MEKIN also allow for the cross section to be a function of the equilibrium xenon concentration. RAMONA-3B is the only code in which the cross sections may depend on boron concentration - essential for considering ATWS events. Although the cross section may also in theory be a function of exposure and void history in RAMONA-3B, the practice at BNL has been to use data that represent the reactor at a fixed point in the fuel cycle.

In a single channel core representation it is necessary to be able to represent more than one bank or group of rods in a dynamic way. Without this capability the rods can only be represented as all in or all out. With this capability (as in BNL-TWIGL and RETRAN-02) the rods are represented with a control rod density between zero and one which varies axially depending on the rod pattern. Having this capability may also be important when the radial dimension is represented in a coarse fashion.

FUTURE TRENDS WITH CORE DYNAMICS CODES

Of the four codes considered for BWR analysis it is clear that the NSSS codes RAMONA-3B and RETRAN-02 have an advantage over the other two codes. The former code can be used for the rod drop accident and both codes can be used for system transients. MEKIN is too long running and does not offer any advantages over RAMONA-3B to be used for BWR calculations. BNL-TWIGL can only be used at BNL and again does not offer any significant advantages over RAMONA-3B. Although these four codes have been highlighted in this paper, work on the advanced code TITAN is progressing and results of that effort will be watched closely to determine whether TITAN or a similar advanced code can replace any of the existing codes.

The major emphasis in the near term with RAMONA-3B and RETRAN-02 is expected to be further code assessment in order to delineate the limitations and capabilities of the codes. Separate effects tests, numerical benchmarks and plant data will be used. This process has already identified deficiencies which can be remedied. The application of the codes to accident analysis is also important as it identifies what reactor behavior is expected and consideration can then be given to whether the modeling for that specific behavior is adequate.

Changes to the core thermal-hydraulics in RAMONA-3B that have already been identified include the improvement of the calculation under reverse flow conditions and the addition of channel dependent fuel rod types for the heat conduction calculation. Naturally, with both RAMONA-3B and RETRAN-02 there will be many modifications that relate to the thermal-hydraulics outside of the core - indeed these problems will be the major preoccupation of future code development. New thermal-hydraulic modeling will also be important for improving computational speed.

Although there has been considerable research in recent years to develop different nodal methods for solving the neutron kinetics equation it is not clear in these times of limited resources if there is sufficient incentive to replace the method used in either code - and in the process give RETRAN-02 a three dimensional neutron kinetics capability. The major motivation would have to be considerably faster computational speed. There is some motivation to enable RAMONA-3B to automatically calculate one dimensional (1D) cross section data from a three dimensional steady state (RAMONA-3B) calculation in order to expedite the use of the code in a 1D mode for certain transient analysis. Analysts interpreting results from both of these codes would benefit by having reactivity edits available.

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